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MARTIN MARIETTA

ORIGINAL

Preliminary Development of an Integrated Approach to the Evaluation of Pressurized Thermal Shock as Applied to the Oconee Unit 1 Nuclear Power Plant

Prepared for the
Division of Risk Analysis and Operations
Office of Nuclear Regulatory Research
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Instrumentation and Controls Division

PRELIMINARY DEVELOPMENT OF AN INTEGRATED APPROACH TO THE EVALUATION OF PRESSURIZED THERMAL SHOCK AS APPLIED TO THE OCONEE UNIT 1 NUCLEAR POWER PLANT

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thermal-hydraulic and fracture-mechanics analysis into an integrated risk analysis methodology, it is not surprising that the results of the thermal-hydraulics and fracture-mechanics calculations are not as closely integrated with each other and with the event-tree analyses as in the subsequent two studies, as described below. As a result, this report should be viewed as the documentation of the initial attempt to address the complex phenomenon termed "pressurized thermal shock" as a coherent whole, with the Oconee study serving as the springboard for the Calvert Cliffs and H. B. Robinson studies in which the methodology was refined and improved.

Several examples of the evolution of the methodology can be given. One is the selection of the cutoff frequency used to "prune" the event trees. For the Oconee Unit 1 study, a frequency value of 1×10^{-6} events per reactor year (1.0E-06/RY) was initially utilized, and all sequences falling below this value were combined to yield a residual risk component. Using a value of 1.0E-06 resulted in the residual component dominating the overall risk measure. Although an attempt to analyze some of the residual sequences was performed (and is incorporated in this report), it is apparent that a smaller value for the cutoff frequency is more appropriate. Therefore, a value of 1.0E-07 was utilized for the Calvert Cliffs and H. B. Robinson studies.

Another example of improvement in methodology is the calculation of the likelihood of an individual sequence driving a crack through the wall. Due to the potential initiation of PTS transients via secondary side upsets (e.g., overcooling transients), much of the thermal-hydraulic efforts pertaining to this study involved the detailed modeling of the secondary side of the nuclear power plants studied. In fact, it is believed that this

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PREFACE

This report summarizes the results of a plant-specific evaluation of pressurized thermal shock (PTS) based on the Oconee Unit 1 Nuclear Power Plant. This evaluation was the first of three undertaken by ORNL for the Nuclear Regulatory Commission. (The other two evaluations were for Calvert Cliffs Unit 1 and H. B. Robinson Unit 2.)

Assessment of the PTS risk to the Oconee plant required the development of a specialized methodology combining various elements of risk analysis, thermal hydraulics, and fracture mechanics. Moreover, since the PTS evaluation was carried out in terms of an overall risk analysis framework, the supporting thermal-hydraulic and fracture-mechanics analyses were required to be probabilistic in nature. The integration of these diverse disciplines into a coherent whole was a major task of this first study. In particular, the development of system state tree methodology, a probabilistic fracture-mechanics model, a simplified thermal-hydraulics approach, and sensitivity and uncertainty analyses as applied to PTS were all introduced in the Oconee analysis (and later refined in the evaluations for Calvert Cliffs and H. B. Robinson).

Development of a new methodology such as that used for the PTS evaluation is evolutionary; that is, initially selected procedures may prove to be not the optimum ones. The philosophy followed on the three PTS pilot studies was to apply the lessons learned from each study to improve the next study, and not go back and reanalyze the first study with improved methods. Since this particular study represented the first attempt to combine both

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study marked the first large-scale application of the computer codes TRAC and RELAP to the secondary side of nuclear steam supply systems. One of the problems of this initial study was that the coolant temperature during a postulated event sequence was calculated in detail only during the initial (rapidly moving) phase of the transient and the temperature/pressure descriptions were then extrapolated to longer times (roughly two hours). Hence, for those transients in which this procedure was utilized to determine the full two-hour transient, the longer term pressures and temperatures are subject to an additional amount of uncertainty. Also a large number of possible sequences were "binned" into relatively few categories of thermal-hydraulic transients for estimating the conditional probability of a through-the-wall crack. In general, this led to conservative estimates of vessel failure for each transient considered. In the Calvert Cliffs and H. B. Robinson studies, thermal-hydraulic and fracture-mechanics calculations were performed for a higher percentage of the transients and there was less reliance on extrapolations.

Owing to the limited number of thermal-hydraulic transients available (either calculated or estimated), the influence of the reactor operator as modeled in the Oconee study by Operator Action Trees was often reduced to a single action (e.g., isolating the feedwater systems), even though it was understood that other actions would also influence the course of a given transient. Due to the limitations, only the operator action regarded as most significant was incorporated into the analysis. In the other two studies an alternate method (i.e., influence diagrams) of accounting for the operator during the course of a PTS transient was employed.

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As part of the methodology development of this study, the sensitivity and uncertainty analyses were approached from an analytical standpoint. The use of analytical procedures to assess the uncertainty in the results for the Oconee study necessitated the use of symmetric distributions to represent the various components of the overall uncertainty (that is, the uncertainty arising from the event tree analyses, the thermal-hydraulic analyses, etc.). In retrospect, the assumption of symmetric uncertainty distributions was not appropriate for many of the parameters addressed, such as the coolant temperatures reached at the end of a two-hour transient. However, since the analytical procedure chosen for the Oconee sensitivity and uncertainty analyses required the use of such distributions, the (probably conservative) assumption that the actual distributions could be approximated by symmetric ones was made. The methodology utilized in the Calvert Cliffs and H. B. Robinson studies was changed to a stochastic or Monte Carlo procedure.

In conclusion, it is the authors' and NRC program monitor's opinion that this report is important because it presents the basic analysis methodology that was later finalized in the later two studies. In addition, it provides insight with respect to the PTS issue for the Oconee Unit 1 plant. However, the reader should be cautioned in the use of the absolute numbers presented in the report.

Thomas J. Burns Principal Author

Carl Johnson NRC Technical Monitor

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ABSTRACT

An evaluation of the risk to the Oconee-1 nuclear plant due to pressurized thermal shock (PTS) has been completed by Oak Ridge National Laboratory (ORNL). This evaluation was part of a Nuclear Regulatory Commission (NRC) program designed to study the PTS risk to three nuclear plants: Oconee-1, a Babcock and Wilcox reactor plant owned and operated by Duke Power Company; Calvert Cliffs-1, a Combustion Engineering reactor plant owned and operated by Baltimore Gas and Electric Company; and H. B. Robinson-2, a Westinghouse reactor plant owned and operated by Carolina Power and Light Company. Studies of Calvert Cliffs-1 and H. B. Robinson-2 are still underway.

The specific objectives of the Oconee-1 study were to: (1) provide a best estimate of the probability of a through-the-wall crack (TWC) occurring in the reactor pressure vessel as a result of PTS; (2) determine dominant accident sequences, plant features, operator and control actions and uncertainty in the PTS risk; and (3) evaluate effectiveness of potential corrective measures. To perform the studies, ORNL constructed millions of hypothetical overcooling events (using computer-generated event trees) and then estimated both the consequences and the probabilities of occurrence for each of these events. A screening frequency of 10^{-6} per reactor year was used to screen out those event tree branches (scenarios) whose frequencies were below that value. All of the remaining scenarios were considered explicitly. The scenarios initially screened out were not discarded but rather were included in a residual category, and their probability contributions were included in the study. Detailed thermal-hydraulic (T-H)

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analyses were performed on a few of the scenarios by Los Alamos National Laboratory and Idaho National Engineering Laboratory and these were reviewed by Brookhaven National Laboratory. For some transients, downcomer mixing calculations were performed by Purdue University. Thermal-hydraulic consequences of all remaining transients (including as one group all those in the residual) were estimated by Science Applications, Inc. For all transients, probabilistic fracture-mechanics calculations were performed by ORNL. The results of all these analyses were integrated by ORNL to predict the probability of TWC for Oconee-1.

Our best estimate of the frequency of a TWC at Oconee-1 due to PTS at 32 EFPY was 4.5 x 10^{-6}/reactor year. However, an uncertainty analysis performed by ORNL for this best-estimate frequency indicates that a factor of about 100 is an appropriate 95% confidence interval, assuming a log-normal uncertainty distribution. Steamline breaks were the most significant contributors to the TWC risk for the types of sequences considered. Uncertainty in downcomer temperatures was the most important contributor to the overall uncertainty in the risk. The most important plant features which were determined to reduce PTS risk at Oconee-1 were: (1) internal vent valves and (2) feedwater pump trip on high steam generator level. The most important operator action for negating PTS was the isolation of a steam generator during a steamline break.

Minimal systems interactions were considered in this study and no external events (fire, floods, seismic events) were considered.

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1. INTRODUCTION

J. D. White, Oak Ridge National Laboratory

1.1. Background

Before the late 1970s it was postulated that the most severe thermal shock a pressurized-water reactor vessel would be required to withstand would occur during a large-break loss-of-coolant accident (LOCA). In this type of overcooling transient, room-temperature emergency core coolant would flood the reactor vessel within a few minutes and rapidly cool the vessel wall. The resulting temperature difference across the wall would cause thermal stresses, with the inside surface of the wall in tension. However, the addition of pressure stresses to the thermal stresses was not considered, since it was expected that during a large-break LOCA the system would remain at low pressure.

In 1978, the occurrence of a non-LOCA-type event at the Rancho Seco Nuclear Power Plant in California showed that during some types of overcooling transients the rapid cooldown could be accompanied by repressurization of the primary system, which would compound the effects of the thermal stresses. As long as the fracture resistance of the reactor vessel remains relatively high, such transients are not expected to cause the reactor vessel to fail. However, after the fracture toughness of the vessel is gradually reduced by neutron irradiation, severe pressurized thermal shock (PTS) might cause a small flaw already existing near the inner surface of

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the wall to propagate through the wall. Depending on the progression of the accident, such a through-the-wall crack (TWC) could lead to core melting.

Following the Rancho Seco incident, the Nuclear Regulatory Commission (NRC) designated pressurized thermal shock as an unresolved safety issue (A-49), and the effects of pressurized thermal shock at operating PWRs were analyzed with input from the owner groups and from eight selected utilities. On the basis of these analyses, NRC concluded that no event having a significant probability of occurring could cause a PWR vessel to fail today or within the next few years. However, NRC projected that as PWR vessels are irradiated, particularly those containing copper in their welds, a few vessels could eventually become susceptible to pressurized thermal shock (SECY-82-465, SECY-83-288, and SECY-83-443).

In order to address the PTS possibility, NRC published a proposed rule that (1) establishes a screening criterion on the reference temperature for nil-ductility transition (RTNDT), (2) requires licensees to accomplish reasonably practicable flux reductions to avoid exceeding the screening criterion, and (3) requires plants that cannot stay below the screening criterion to submit a plant-specific safety analysis to determine what, if any, modifications are necessary if continued operation beyond the screening limit is allowed.

In addition, NRC organized a PTS research project, described in part in this report, to help confirm the technical bases for the proposed PTS rule and to aid in the development of guidance for the licensee plant-specific

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PTS analyses, as well as the development of acceptance criteria for proposed corrective measures. The research project consisted of PTS pilot analyses for three PWRs: Oconee Unit 1, designed by Babcock and Wilcox; Calvert Cliffs Unit 1, designed by Combustion Engineering; and H. B. Robinson Unit 2, designed by Westinghouse. The study team consisted of Oak Ridge National Laboratory (ORNL), Idaho National Engineering Laboratory (INEL), Los Alamos National Laboratory (LANL), Brookhaven National Laboratory (BNL), and Purdue University, with the results being integrated by ORNL. The results of the first of the three planned pilot analyses, that for Oconee Unit 1, is described in this report. The second analysis, for Calvert Cliffs Unit 1, and the third, for H. B. Robinson Unit 2, are described in separate reports.[1,2]

1.2. Overall Objectives of PTS Studies

The overall objectives of the PTS studies at ORNL are: (1) to provide for each of the three plants an estimate of the probability of a crack propagating through the wall of a reactor pressure vessel due to pressurized thermal shock, (2) to determine the dominant overcooling sequences, plant features, and operator and control actions and the uncertainty in the plant risk due to pressurized thermal shock; and (3) to evaluate the effectiveness of potential corrective measures. ORNL was also to determine what parts of the studies might have generic applicability.

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1.3. Limitations of the Studies

Determining the consequences of a through-the-wall crack was not a part of the program; that is, studies of the geometry of a through-the-wall crack, missile formation, the means for cooling the core, the extent of radiation releases, and risks to the public were not addressed. These consequences are to be studied under other NRC-sponsored work.

Neither did the program consider the effects of external events, such as earthquakes, fires, and floods (both external and internal to the containment), and sabotage. ORNL suspects that the effect of excluding such events is not serious because of (1) the low probabilities that the events will occur and (2) the likelihood that failures of systems due to external events would cause undercooling situations rather than overcooling situations.

1.4. PTS Analysis for Oconee Unit 1

This report describes the PTS analysis of Oconee Unit 1, a PWR designed by Babcock and Wilcox (B&W). The reactor is owned and operated by the Duke Power Company.

The reactor coolant system of Oconee Unit 1 has two hot legs and four cold legs and utilizes two B&W once-through steam generators. The PTS analysis for the unit consisted of:

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(1) gathering plant data,

(2) building event tree models and thermal-hydraulic models,

(3) quantifying frequencies of event tree end states,

(4) predicting thermal-hydraulic responses of the plant to the events,

(5) calculating the conditional probability of a through-the-wall crack (TWC) for each event,

(6) integrating steps 3 and 5 to produce an estimate of overall through-the-wall crack frequency at Calvert Cliffs Unit 1 due to all events considered,

(7) performing sensitivity and uncertainty analyses on the results, and

(8) evaluating potential corrective measures.

In support of the program, Duke Power Company provided the research team with copies of plant drawings, plant data and operating procedures for Oconee Unit 1. Thermal-hydraulic analysis models were developed by Idaho National Engineering Laboratory (INEL), Los Alamos National Laboratory (LANL), Science Applications Incorporated (SAI) under subcontract to ORNL, and Purdue University under other NRC-funded programs supporting the ORNL PTS studies.

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The SAI models were used by ORNL to provide an early understanding of plant behavior, which was useful in building event tree models. The models developed by INEL and LANL, which became available later, were necessarily very complicated and expensive to run, and because of funding limitations, only a few overcooling scenarios (less than 20) could be calculated explicitly. The scenarios calculated were chosen carefully to provide a basis for estimating the results of the remaining thousands of potential scenarios.

INEL and LANL calculations were reviewed by Brookhaven National Laboratory. In addition, because the computer code used by INEL (the RELAP5 code) and LANL (the TRAC code) could not perform detailed mixing calculations, additional thermal-hydraulic studies that included mixing were performed under another NRC contract by Purdue University. These detailed analyses by INEL, LANL and Purdue were then used by SAI as a basis for estimating the Oconee Unit 1 thermal-hydraulic behavior during the remainder of the hypothetical transients.

1.5. Description of This Report

This report presents the results of the specific study for Oconee Unit 1 and describes the methodology developed for performing the analysis. Chapter 2 describes the plant's components and operational behavior characteristics that are believed by ORNL to be pertinent to the PTS issue.

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Hopefully, this chapter and the accompanying references could be used to build other models of the unit. The reader is advised, however, that building a model useful in PTS studies is a difficult process due to the many complex systems interactions occurring in operational upsets and the model may not be applicable to other types of transients.

Chapter 3 describes the hypothetical overcooling sequences considered in the analysis. The methodology used to determine what sequences are possible and how frequencies for the sequences are estimated is discussed in detail. An event tree approach was chosen; no fault trees were used in this analysis.

Chapter 4 discusses the thermal-hydraulics models and summarizes the calculations from the INEL, LANL, SAI and Purdue analyses. From this chapter, the reader can obtain a good understanding of how Oconee Unit 1 is predicted to behave under hypothetical overcooling scenarios.

Chapter 5 describes the calculations of conditional TWC probabilities for groups of thermal-hydraulic responses. This work, done at ORNL, utilized probabilistic fracture-mechanics analytical methods in assessments of the probability that cracks might propagate through the reactor vessel wall. The chapter describes the vessel welds and their chemistries and gives estimated fluences throughout the expected plant lifetime. The assumed crack densities and distributions are also described.

The integration of the event sequences analysis, thermal-hydraulic analysis and fracture-mechanics analysis to produce an overall best estimate of PTS

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risk at Oconee Unit 1 is described in Chapter 6. In this chapter the dominant contributions to the risk and effects of potential corrective measures are discussed. Although a need for corrective measures at Oconee Unit 1 has not been established, the effects of corrective measures were studied to give the NRC or other future analysts an idea of the relative importance of different corrective actions. The overall effects of PTS corrective measures on plant safety and their cost effectiveness have not been examined.

An analysis of the uncertainties performed by SAI and ORNL is described in Chapter 7, in which the major contributors to the uncertainty in overall PTS risks are identified.

Conclusions of the study and recommendations are given in Chapter 8. It should be noted that Appendix G is not referred to in the text but represents a series of utility comments and the authors' responses to those comments.

REFERENCES

1. D. L. Selby, et al., Pressurized Thermal Shock Evaulation of Calvert Cliffs Unit 1 Nuclear Power Plant, NUREG/CR-4022 (ORNL/TM-9408), September 1985.

2. D. L. Selby, et al., Pressurized Thermal Shock Evaulation of H. B. Robinson Unit 2 Nuclear Power Plant, NUREG/CR-4183 (ORNL/TM-9567), September 1985.

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2.0 PLANT DESIGNS AND OPERATIONS PERTINENT TO PRESSURIZED THERMAL SHOCK

L. B. LaMonica, Science Applications, Incorporated

2.1 Introduction

The elements working together in PTS situations include the physical design of plant systems, actions of automatic controls, and operator intervention. These elements influence both thermal-hydraulic and probabilistic analyses of overcooling events. This section describes only the hardware systems, controls, and operations pertinent to PTS risk evaluation. A complete description of the Oconee-1 plant is beyond the scope of this document. This section includes summary descriptions of the reactor cooling system (RCS) primary and the feedwater, steam generator, and steam distribution (secondary) systems. The physical relationships of these systems as they bear on PTS is addressed. More general discussions of Oconee-1 systems may be obtained from the Oconee FSAR[2.1] and other sources.[2.2,2.3] The instrumentation and controls influencing the response of the primary and secondary loops and the plant auxiliaries are discussed in Section 2.2.

Section 2.3 addresses the intended operation of hardware and controls in conjunction with operator actions as defined by plant procedures.

2.2 Systems

2.2.1 Reactor Cooling Systems (RCS)

The Oconee-1 Nuclear Power Station employs a Babcock & Wilcox (B&W) designed 177 fuel assembly pressurized water reactor (PWR) rated at 2568 MWt. The two cooling loops, A and B, consist of a hot leg, a once-through steam

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generator (OTSG), two reactor coolant pumps (RCP) per loop, and two cold legs per loop. The unit is of lowered loop design, meaning the bases of the steam generators lie below the horizontal plane through the top of the reactor core. This arrangement is illustrated in Figure 2.1. The cold legs discharge coolant into an annular downcomer between the vessel wall and the thermal shield (Figure 2.2). The coolant flows down to the lower plenum and then up through the core region to the upper plenum and into the hot legs. The pressures, temperatures, and mass flows for this system at hot standby and full power conditions are summarized in Table 2.1.

A pressurizer vessel connected to loop A provides pressure control for the primary coolant system. The coolant inventory control system augments the pressure control function by adding or withdrawing coolant based on indicated inventory.

The emergency core cooling systems (ECCS) include high-pressure injection (HPI), core flood tanks (CFT), and low-pressure injection systems (LPI). These safety grade systems inject cool water into the primary to prevent fuel overheating under accident conditions. The HPI system can also repressurize the RCS for certain small-break loss-of-coolant accident (LOCA) and overcooling events and so is an important factor in PTS evaluation.

The features of the different reactor cooling system components are described below.

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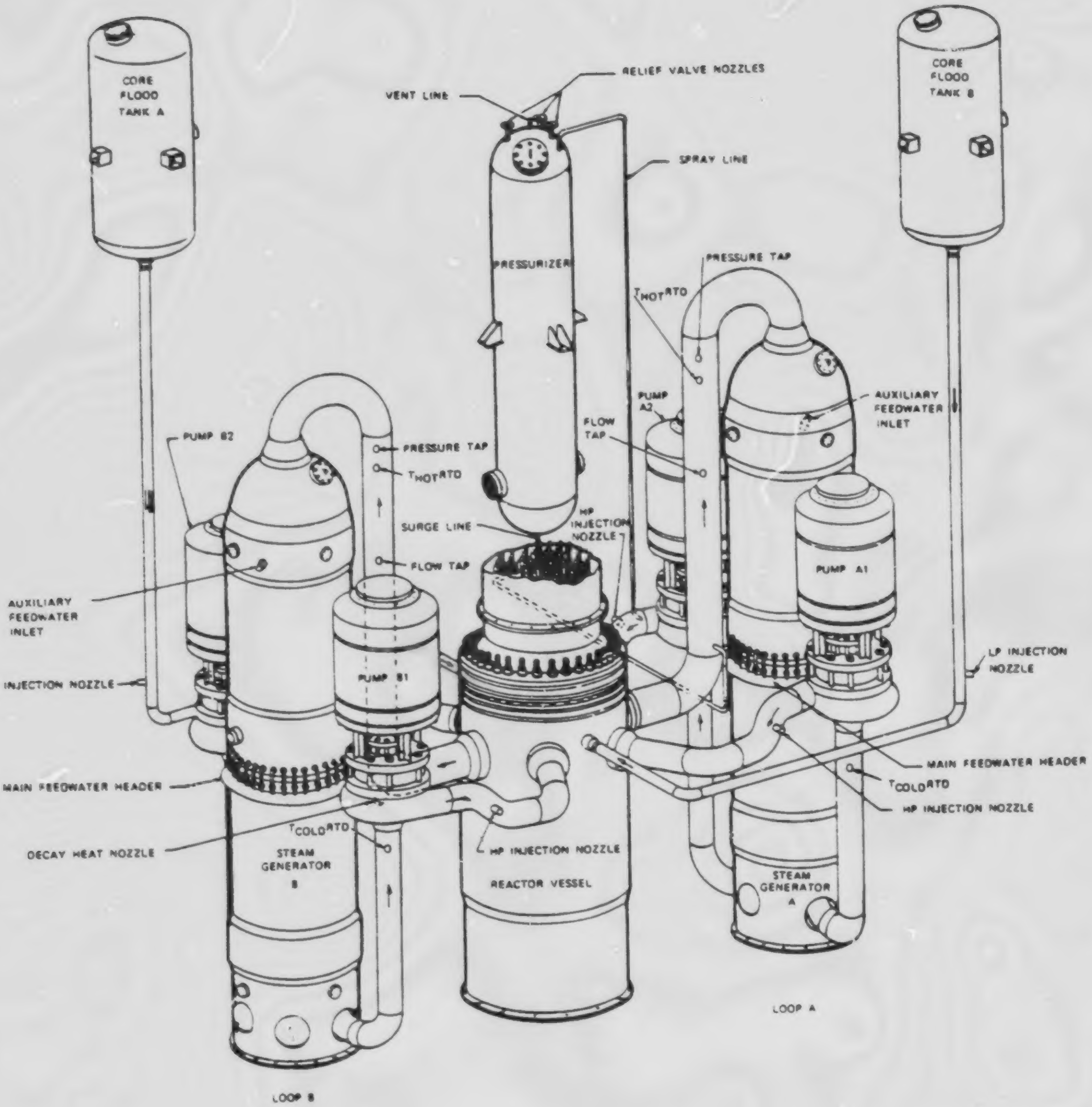
CORE FLOOD TANK A
VENT LINE
RELIEF VALVE NOZZLES
CORE FLOOD TANK B
SPRAY LINE
PRESSURIZER
PRESSURE TAP
T_{HOT}RTD
PUMP A2
FLOW TAP
AUXILIARY FEEDWATER INLET
PUMP B2
PRESSURE TAP
T_{HOT}RTD
SURGE LINE
HP INJECTION NOZZLE
FLOW TAP
PUMP A1
AUXILIARY FEEDWATER INLET
PUMP B1
LP INJECTION NOZZLE
LP INJECTION NOZZLE
MAIN FEEDWATER HEADER
MAIN FEEDWATER HEADER
T_{COLD}RTD
HP INJECTION NOZZLE
DECAY HEAT NOZZLE
T_{COLD}RTD
STEAM GENERATOR B
HP INJECTION NOZZLE
REACTOR VESSEL
STEAM GENERATOR A
LOOP A
LOOP B

Figure 2.1 Oconee-1 PWR coolant system.

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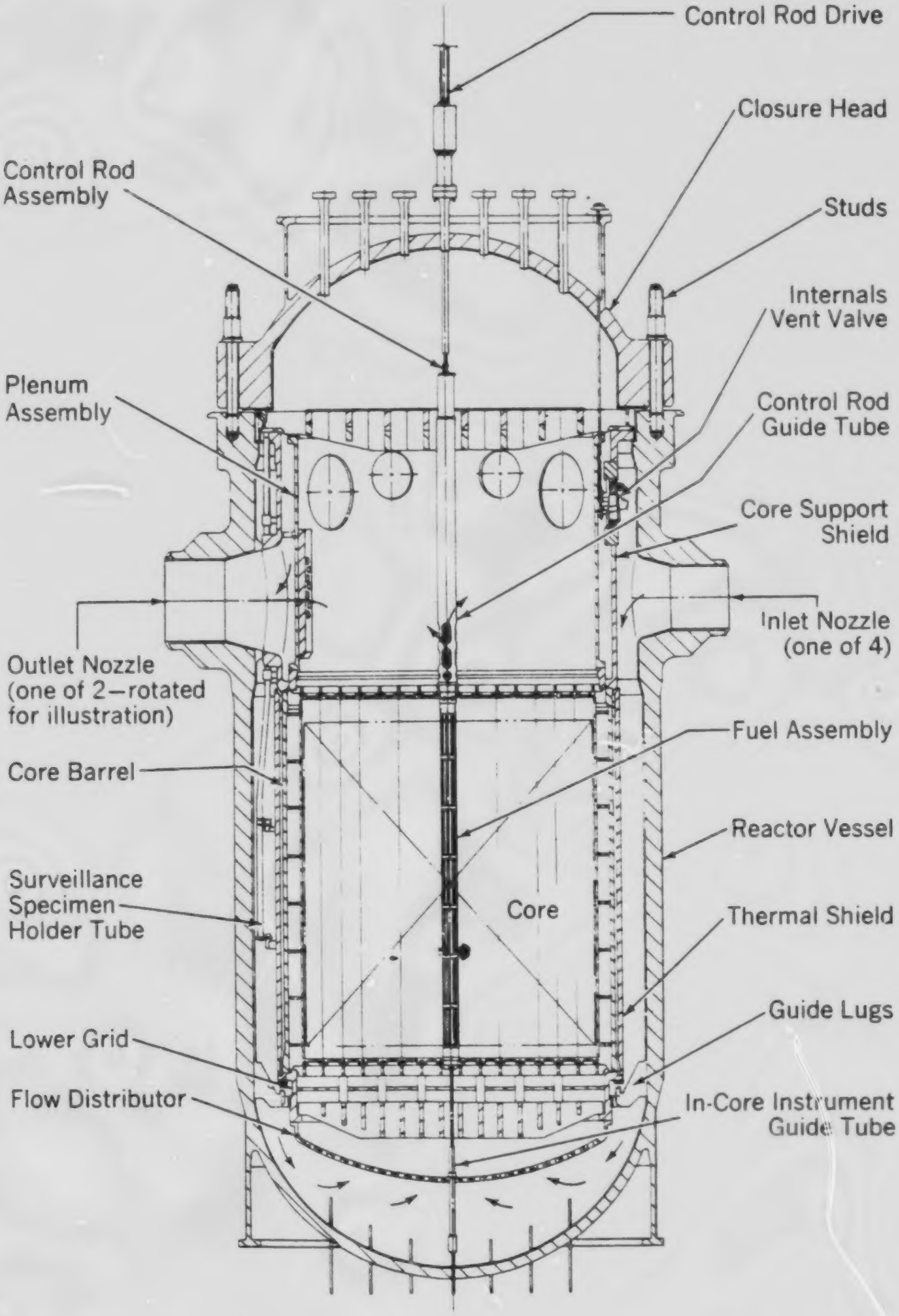
Control Rod Drive
Closure Head
Control Rod Assembly
Studs
Internals Vent Valve
Plenum Assembly
Control Rod Guide Tube
Core Support Shield
Inlet Nozzle (one of 4)
Outlet Nozzle (one of 2—rotated for illustration)
Fuel Assembly
Core Barrel
Reactor Vessel
Surveillance Specimen Holder Tube
Core
Thermal Shield
Guide Lugs
Lower Grid
Flow Distributor
In-Core Instrument Guide Tube

Figure 2.2 Detail of Oconee-1 PWR vessel.

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Table 2.1 Oconee-I PWR steady state conditions[2.4,2.5]

Parameter	Full Power	Hot Zero Power
RCS Primary		
Power (MW)	2568.0	9.0
Coolant flow rate, total (Kg/s)	17640.0	
Hot leg temperature (K)	589.3	
Cold leg temperature (K)	563.5	551.0
Primary pressure (MPa)	14.96	14.8
Core pressure drop (MPa)	0.11	
Vessel pressure drop (MPa)	0.41	
Pressurizer water level (m)	5.6	
HPI/LPI coolant temperature (K)	305.4	305.4
Accumulator coolant temperature	305.4	305.4
Steam Generator Secondary		
Feedwater flow per loop (Kg/sec)	680.4	
Feedwater temperature (K)	511.0	305.4
Steam generator outlet pressure (Mpa)	6.38	6.21
Steam outlet superheat (K)	33.3	0.0
Steam generator secondary inventory (Kg)	<1.77E4	<2.64E4
Steam generator nominal level (m)	6.1 (operating)	9.1 (startup)
MFW pump inlet pressure (MPa)	2.77	
MFW pump inlet temperature (K)	461.0	305.4
C.B. pump inlet pressure (MPa)	0.8	
C.B. pump inlet temperature (K)	308.3	305.4
Hotwell pressure (MPa)	0.01	0.01
Hotwell temperature (K)	305.4	305.4
Hotwell inventory (Kg)	5.31E5	5.31E5
Upper surge tank inventory (Kg)	2.69E5	2.69E5

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2.2.1.1 Reactor Vessel

The reactor pressure vessel is illustrated in Figure 2.2. The core barrel divides the volume of the vessel into a core region and an annular region called the downcomer. The core barrel provides structural support for the core and defines the coolant flow path. Coolant enters the downcomer annulus through the four cold leg nozzles located on an elevation above the top of the core. The two hot leg nozzles pass through the downcomer region and the core barrel and have the same center-line elevation as the cold leg nozzles.

There are eight vent valves in the core barrel above the hot and cold leg nozzles. The primary purpose of the vent valves is to facilitate core reflooding following a postulated large-break LOCA in the reactor cold leg piping by allowing flow of steam from the upper plenum to the break. The vent valves also allow flow of hot water into the downcomer region during PTS transients when natural circulation flow is restricted or lost. The hot vent valve flow mixes with the HPI flow to moderate downcomer temperatures.

Each vent valve consists of a hinged disc and valve body situated to remain closed under normal operating conditions (i.e., with RCP running). When natural circulation and/or flow stagnation is experienced, the differential pressure drop between the downcomer and upper plenum reverses, causing the valves to open. The valves begin to open on a differential pressure of 0.85 KPa (0.125 psid) and are fully open at 1.79 KPa (0.26 psid).

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Besides these hydraulic characteristics, the vessel embodies metallurgical and neutron fluence factors important to PTS. These are discussed in Chapter 5.

2.2.1.2 Steam Generators and Primary Piping

The configuration of the once-through steam generators is depicted in Figure 2.3. In this configuration the primary coolant makes a single pass from top to bottom through straight tubes. Steam generated outside the tubes flows up the shell side of the steam generator, countercurrent to primary coolant downward flow. This arrangement promotes superheating of steam during normal operations. During transient operation, primary reheating (reverse heat transfer) by steam condensation on the tubes may contribute to loop flow stagnation. The secondary side of the steam generators is discussed in Section 2.2.2.

The hot leg pipes connecting the vessel outlet to the upper plenum of each steam generator are referred to as "candy canes" due to their appearance (see Figure 2.1). Under natural circulation conditions, steam voids may collect at the top of these pipes and block natural circulation flow in the loop.

Coolant exits the tubes at the lower plenum of the steam generator. The loop flow is split between the two cold legs at the lower plenum of the steam generator (Figures 2.1 and 2.3). The piping forms a trap or loop seal at its lowest point and then runs upward to the reactor coolant pumps at the same level as the vessel inlet-outlet nozzles.

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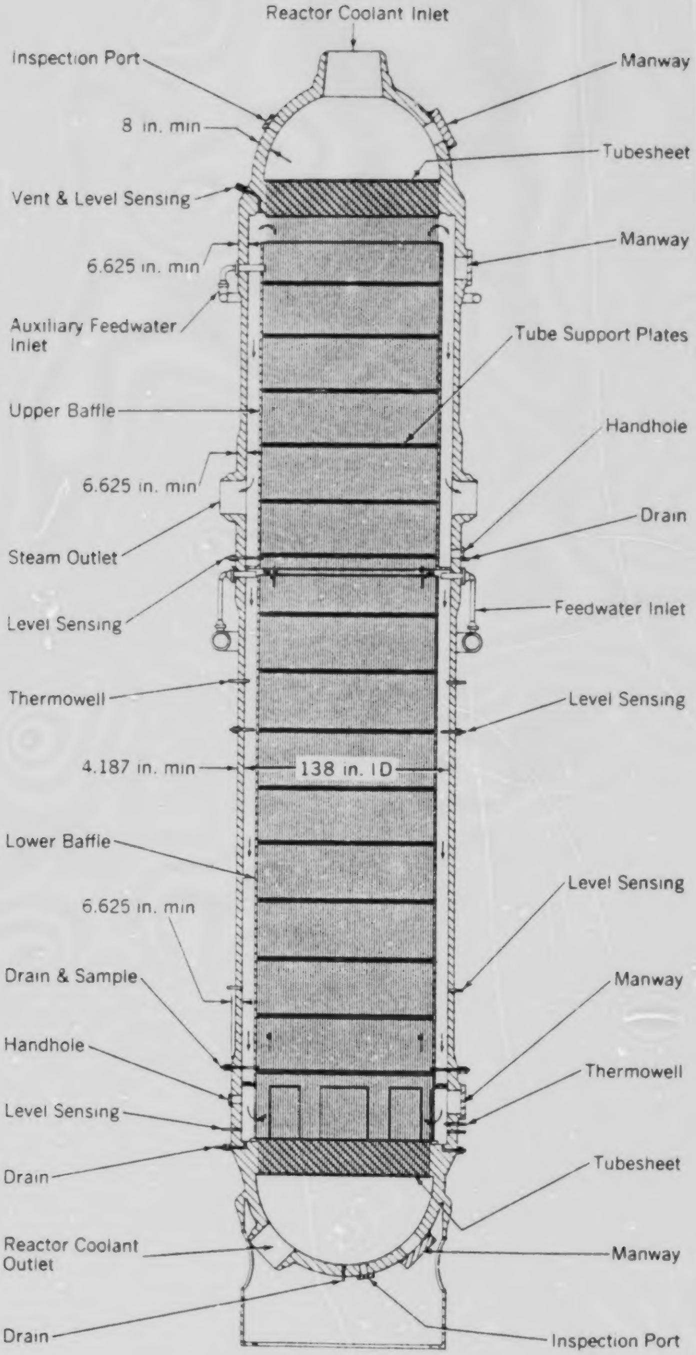
Reactor Coolant Inlet
Inspection Port
Manway
8 in. min
Tubesheet
Vent & Level Sensing
Manway
6.625 in. min
Auxiliary Feedwater Inlet
Tube Support Plates
Upper Baffle
Handhole
6.625 in. min
Drain
Steam Outlet
Level Sensing
Feedwater Inlet
Thermowell
Level Sensing
4.187 in. min
138 in. ID
Lower Baffle
Level Sensing
6.625 in. min
Drain & Sample
Manway
Handhole
Thermowell
Level Sensing
Drain
Tubesheet
Reactor Coolant Outlet
Manway
Drain
Inspection Port

Figure 2.3 Detail of once through steam generator.

2.2.1.3 Reactor Coolant Pumps

Forced circulation is maintained by four reactor coolant pumps (RCP), two on each loop. Each pump has a rated flow of 5.56 m^3/s (88,000 gpm) with a developed head of 0.78 MPa (113 psid). The rated maximum pump power is 6.7 MW (9000 hp) per pump. The pump flywheel inertia is sufficient to prolong pump spindown for about 100 s after trip, maintaining forced circulation for the same period.

2.2.1.4 Pressurizer and Pressure Control

The pressurizer, depicted in Figure 2.4, serves as an expansion or surge tank for controlling system pressure. A 0.254m (10-in.) i.d. pipe connects the pressurizer to the loop A hot leg. During normal operation the pressurizer contains 19.8m^3 (700 ft^3) of steam and 22.7m^3 (800 ft^3) of water at saturation conditions. The pressurizer heater has a maximum input capacity of 1.638 MW. Rapid overcooling incidents usually lower the water level below the heater trip level, removing the heater from operation. Unless it is manually reset, the heater has little influence on pressure recovery for overcooling events.

The pressurizer is also equipped with one power-operated relief valve (PORV) and two code safety relief valves. A motor-operated block valve can be used to isolat the PORV. The PORV has an effective inside diameter of 2.77 cm (1.09 in.) and a design steam flow capacity of 13.51 kg/s (107,000 lb/h). The PORV is programmed to open at 16.9 MPa (2450 psig) and close at 16.6 MPa (2400 psig). The two code safety relief valves each have an effective inside diameter of 4.57 cm (1.8 in.) and together have a design

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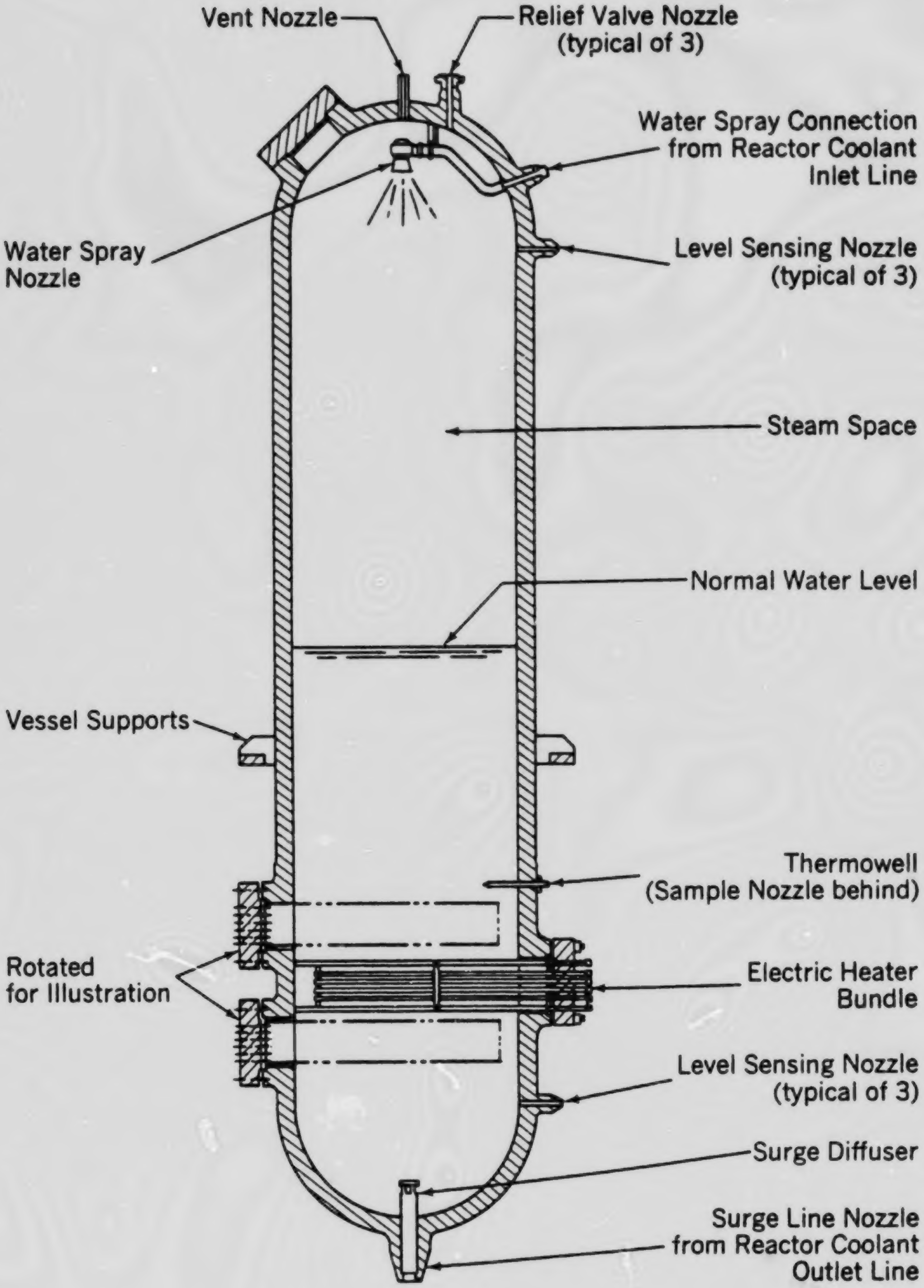
Vent Nozzle
Relief Valve Nozzle (typical of 3)
Water Spray Connection from Reactor Coolant Inlet Line
Water Spray Nozzle
Level Sensing Nozzle (typical of 3)
Steam Space
Normal Water Level
Vessel Supports
Thermowell (Sample Nozzle behind)
Rotated for Illustration
Electric Heater Bundle
Level Sensing Nozzle (typical of 3)
Surge Diffuser
Surge Line Nozzle from Reactor Coolant Outlet Line

Figure 2.4 Detail of pressurizer.

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steam flow capacity of 84.2 kg/s (667,000 lb/h). The code safety relief valves begin opening at a system pressure of 17.2 MPa (2,500 psig).

2.2.1.5 Emergency Core Cooling Systems

High-Pressure Injection System

The high-pressure injection (HPI) system consists of three motor-driven centrifugal pumps and a header system that can deliver flow to each of the cold legs downstream of the respective reactor coolant pumps. During normal operation, one pump in the HPI system runs continuously as part of the inventory control system, injecting makeup water into the loop A cold legs to counter the letdown flow.

When transient events cause depressurization of the primary system to below 10.34 MPa (1500 psig), the safety injection actuation signal starts the two remaining pumps and aligns valves to provide a path for pumping borated water from the borated water storage tank into all four cold legs. Approximately two-thirds of the total flow enters loop A and the remainder enters loop B. Since the HPI system uses centrifugal pumps, the total flow rate is a function of cooling system pressure. This relationship is presented in Figure 2.5. The HPI system can repressurize an intact primary coolant system up to the safety relief valve set point (2500 psig) and can partially repressurize the system during certain small-break LOCA events. After actuation by the safety injection signal, the HPI system remains in operation until it is manually shut off.

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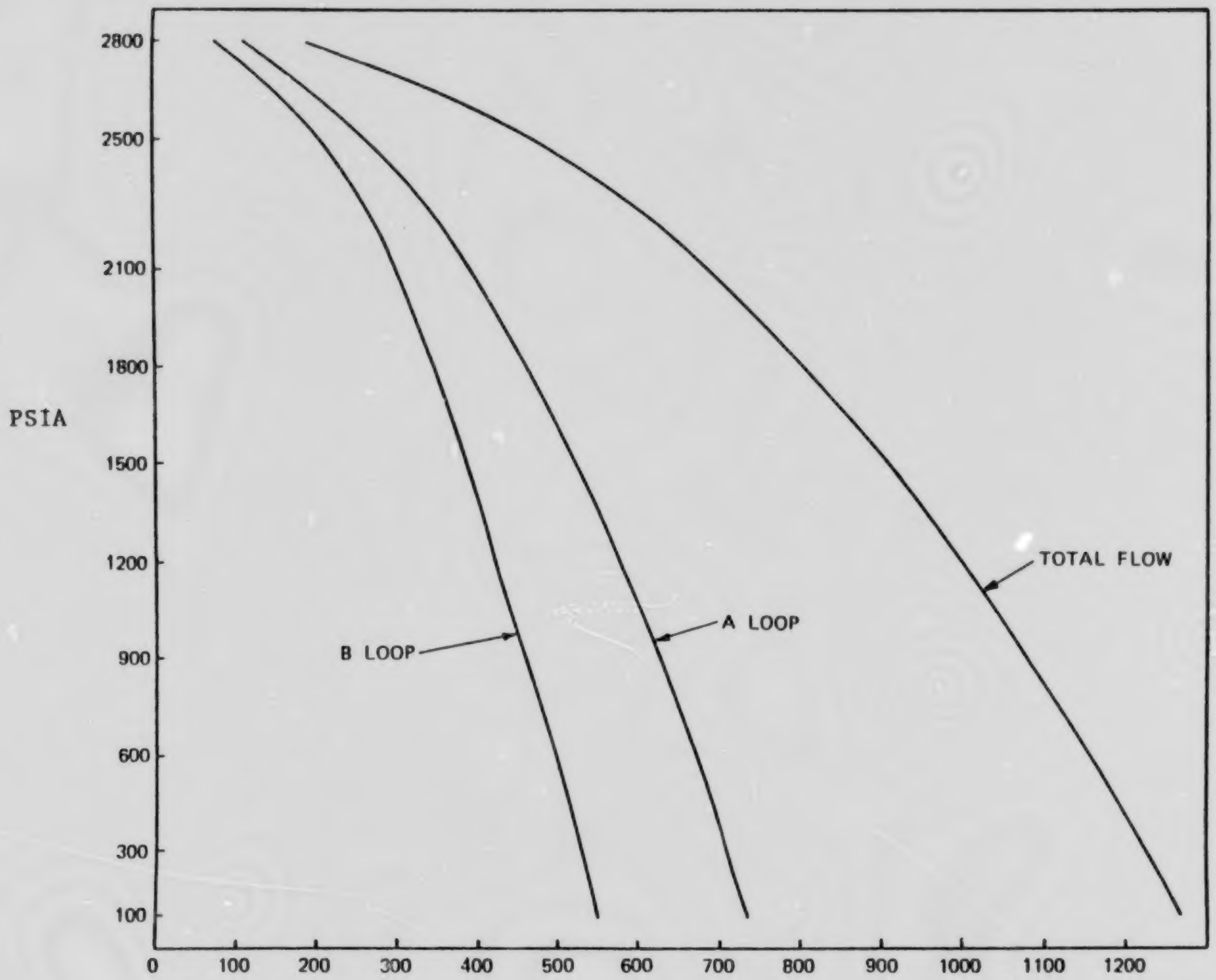

Figure 2.5 High pressure injection system flow capacity.

Core Flood Tanks

The two core flood tanks (CFT) store a total of 59 m^3 (15,560 gal) of borated water under a nitrogen gas blanket at 4.14 MPa (600 psig). If a LOCA or a severe overcooling incident reduces the RCS pressure below the pressure in the core flood tanks, check valves in the 0.356m- (14-in.) i.d. CFT discharge lines will open and the cool, borated water will begin flowing into the reactor vessel downcomer at the two core flood nozzles located 2 ft above the plane of the reactor inlet-outlet nozzles and diametrically opposed to each other (see Figure 2.1). Continued decreases in RCS pressure are necessary to sustain flow from the CFTs into the vessel.

Low-Pressure Injection

When the RCS pressure drops below 3.45 MPa (500 psig), the low-pressure injection (LPI) pumps start and the pump suction and discharge valves are realigned for injection of water from the borated water storage tank into the reactor vessel downcomer at the core flood nozzles. There is no LPI flow until the RCS pressure drops to 1.38 MPa (200 psig), the LPI pump shutoff pressure. At 0.69 MPa (100 psig), a total of 0.379 m^3/s (6,000 gpm) is delivered by LPI. By further realignment of pump suction and discharge valves, the LPI system may also function in the residual heat removal system and other functions not related to PTS. Due to the low primary pressures present when the LPI system is operating, this system will not make any significant contribution to PTS risk.

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2.2.2 Feedwater and Steam Systems

The steam generators and their associated feedwater system and pressure control systems play a pivotal role in PTS sequences. The steam generators provide cooling to the reactor coolant system during both normal operations and overcooling events. The performance of the various feedwater and steam system components influences the rate of cooldown during an accident. The feedwater and steam systems form a closed loop as illustrated in Figure 2.6. The portions of the loop most significant to PTS events are indicated by bold flow lines. The main feedwater (MFW) train runs from the condenser hotwell to the steam generators and supplies a regulated flow of preheated water. The emergency feedwater (EFW) train (not depicted on Figure 2.6) pumps cool water from the condenser hotwell and/or the upper (condensate) surge tank directly to the steam generators without any preheating.

The two once-through steam generators are capable of generating 2.45×10^6 kg/h (5.39×10^6 lb/h) of steam at 6.27 MPa (910 psia) and 572 K (570^{o}F). The once-through steam generator design yields steam superheats of 33.3 ^{o}C (60^{o}F) to improve power cycle efficiency. Achievement of these steam conditions requires precise control of feedwater flows via the integrated control system (ICS).

The main steam lines connect the steam generators to the high-pressure turbine header at the turbine stop valves. A scoping study of the steam and power conversion systems[2.6] suggests that the turbine stop valves serve as the boundary point for PTS modeling of the Oconee-1 plant. Failure of a stop valve will yield no impact if the turbine control valves operate properly or will yield conditions similar to a small or large steam

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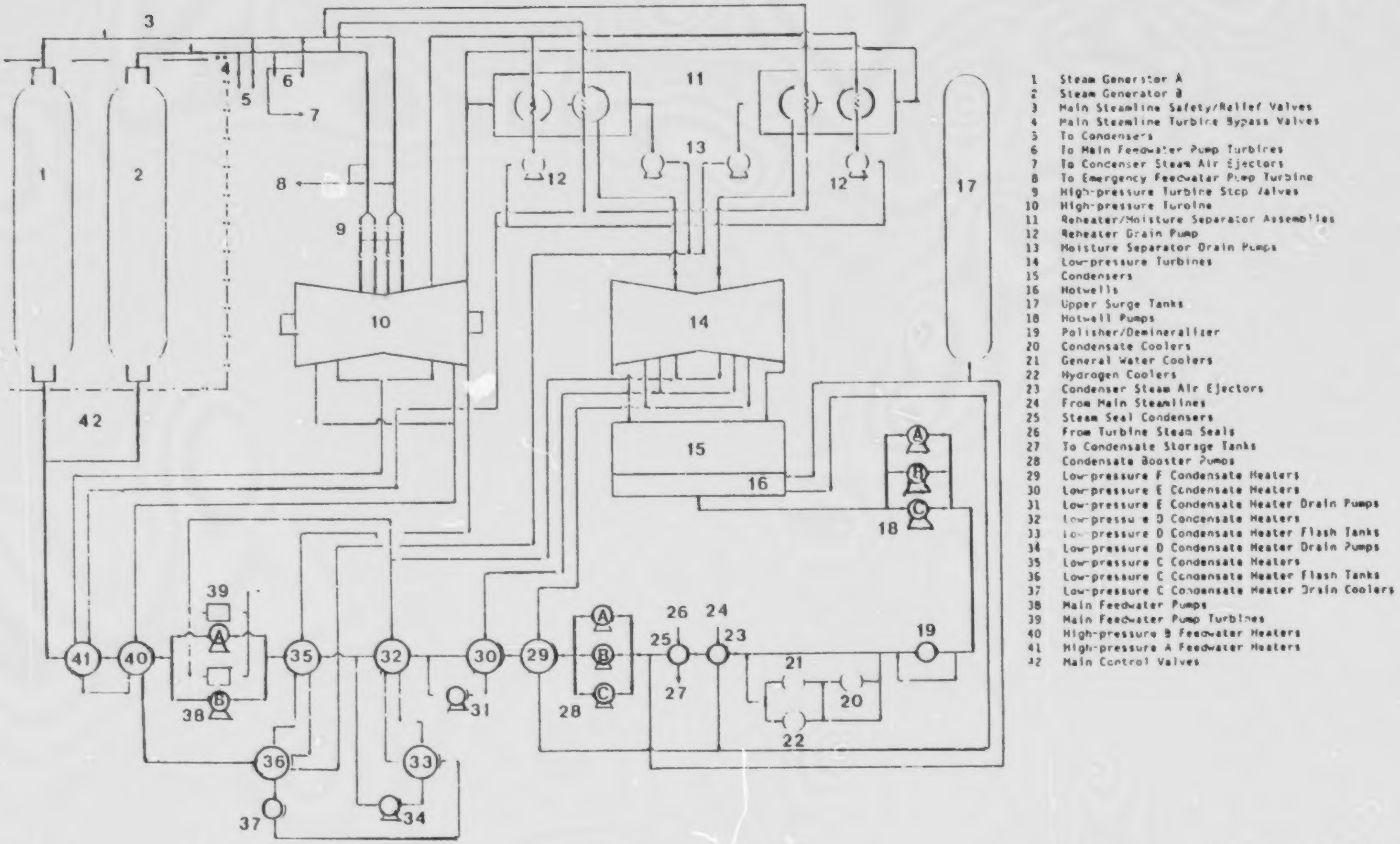
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34
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36
37
38
39
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42
A
B
C
1 Steam Generator A
2 Steam Generator B
3 Main Steamline Safety/Relief Valves
4 Main Steamline Turbine Bypass Valves
5 To Condensers
6 To Main Feedwater Pump Turbines
7 To Condenser Steam Air Ejectors
8 To Emergency Feedwater Pump Turbine
9 High-pressure Turbine Stop Valves
10 High-pressure Turbine
11 Reheater/Moisture Separator Assemblies
12 Reheater Drain Pump
13 Moisture Separator Drain Pumps
14 Low-pressure Turbines
15 Condensers
16 Hotwells
17 Upper Surge Tanks
18 Hotwell Pumps
19 Polisher/Demineralizer
20 Condensate Coolers
21 General Water Coolers
22 Hydrogen Coolers
23 Condenser Steam Air Ejectors
24 From Main Steamlines
25 Steam Seal Condensers
26 From Turbine Steam Seals
27 To Condensate Storage Tanks
28 Condensate Booster Pumps
29 Low-pressure F Condensate Heaters
30 Low-pressure E Condensate Heaters
31 Low-pressure E Condensate Heater Drain Pumps
32 Low-pressure D Condensate Heaters
33 Low-pressure D Condensate Heater Flash Tanks
34 Low-pressure D Condensate Heater Drain Pumps
35 Low-pressure C Condensate Heaters
36 Low-pressure C Condensate Heater Flash Tanks
37 Low-pressure C Condensate Heater Drain Coolers
38 Main Feedwater Pumps
39 Main Feedwater Pump Turbines
40 High-pressure B Feedwater Heaters
41 High-pressure A Feedwater Heaters
42 Main Control Valves

Figure 2.6 Oconee-1 main feedwater, steam, and power systems.

line break if the turbine control valves fail. The operation of the turbines, being downstream from the stop valves, bears no direct impact on PTS. Feedwater heating steam from the numerous extraction lines is lost immediately after turbine trip and does not require extensive modeling other than to determine the temperature decay for the MFW flow. The pertinent characteristics of the main feedwater, emergency feedwater, steam generator, and main steam line systems are discussed in detail below.

2.2.2.1 Main Feedwater Train

The main feedwater train (MFW) or main condensate/feedwater train is schematically illustrated in Figure 2.6. The prime function of the main condensate/feedwater train is to transport subcooled water from the hotwell outlets to the steam generator main feedwater inlets while both pressurizing and heating it. A second obvious function of this train is to control the quantity of feedwater reaching the secondary side of each steam generator. The control function is implemented by the integrated control system (ICS), which is described in Section 2.2.3.

The main condensate/feedwater train begins with low-pressure, low-temperature subcooled condensate exiting the hotwells. The condensate is immediately pressurized by three parallel motor-driven hotwell pumps. These pumps together have a rated flow of 0.8 m^3/s (12,678 gpm) at a differential pressure of 1.152 MPa (167 psi). The pressure rise across the pumps at zero flow would be 1.42 MPa (206 psi). The hotwell pumps will trip on low level in the hotwell and on loss of condenser vacuum.

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The condensate then flows through the polisher/demineralizers, resin trap, and the tube sides of the condensate coolers, hydrogen coolers, general water coolers (these coolers are known collectively as the plant heat dump), condensate steam air ejector, and steam seal condensers. The slightly warmer, low-pressure subcooled condensate leaving the steam seal condensers is then pressurized by three parallel motor-driven condensate booster pumps whose function is to provide adequate suction pressure to the main feedwater pumps after the frictional losses of the low-pressure condensate heaters. The differential pressure rise generated by the condensate booster pumps is 2.94 MPa (426 psi) at rated flow. At zero flow the pressure rise across these pumps would be 3.14 MPa (455 psi). The condensate booster pumps will trip on sustained low suction pressure of 0.21 MPa (30.7 psi).

Departing from the condensate booster pumps, the condensate is heated significantly by passage through the tube side of four stages of low-pressure condensate heaters. The first stage consists of three parallel low-pressure condensate heaters (F heaters); the second stage consists of two parallel low-pressure condensate heaters (E heaters); the third stage consists of two parallel low-pressure condensate heaters (D heaters) in series with two parallel low-pressure condensate heaters (C heaters). The condensate next travels to the two parallel turbine-driven main feedwater pumps that provide final pressurization of the feedwater. Frictional losses in the high-pressure A and B feedwater heaters and main control valves reduce the feedwater pressure to the desired delivery pressure at the steam generator main feedwater inlets. The integrated control system (ICS) varies both the main feedwater pump speed and feedwater control valve position to maintain optimum control of flow rate and pressure drop. The ICS

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is discussed in Section 2.2.3. The main feedwater pumps will trip on high discharge pressure of 8.89 MPa (1289.7 psi), low suction pressure of 1.72 MPa (249.7 psia), and high steam generator level. Should the MFW pumps trip, the hotwell and condensate booster pumps would be capable of delivering flow to the steam generators at pressures up to about 3.80 MPa (550 psig). At higher pressures no flow would be delivered.

Upon exiting the common main feedwater pump discharge header, the subcooled feedwater is further heated to the desired delivery temperature, 511 K (460°F) by two banks of high-pressure feedwater heaters in series (B heaters and A heaters). The feedwater recombines in the high-pressure A feedwater heater outlet header before dividing into two lines containing the main control valves and arriving at the steam generator main feedwater headers. Each main control valve has a startup control valve in parallel (Figure 2.7). The startup control valve is used for low-power conditions until feedwater control can be accomplished with the main control valve. At Oconee, there is no automatic feedwater isolation function separate from the main and startup flow control valves. However, the operator can isolate feedwater using available block valves.

There is a cross-connect from the main feedwater train to the emergency feedwater train. This cross-connect is activated by the feedwater realignment trip which follows RCP trip and resetting of the steam generator level controls. The feedwater realignment valves redirect the startup control valve flow away from the main feedwater header to the emergency feedwater header. Therefore, heated water is made available for filling

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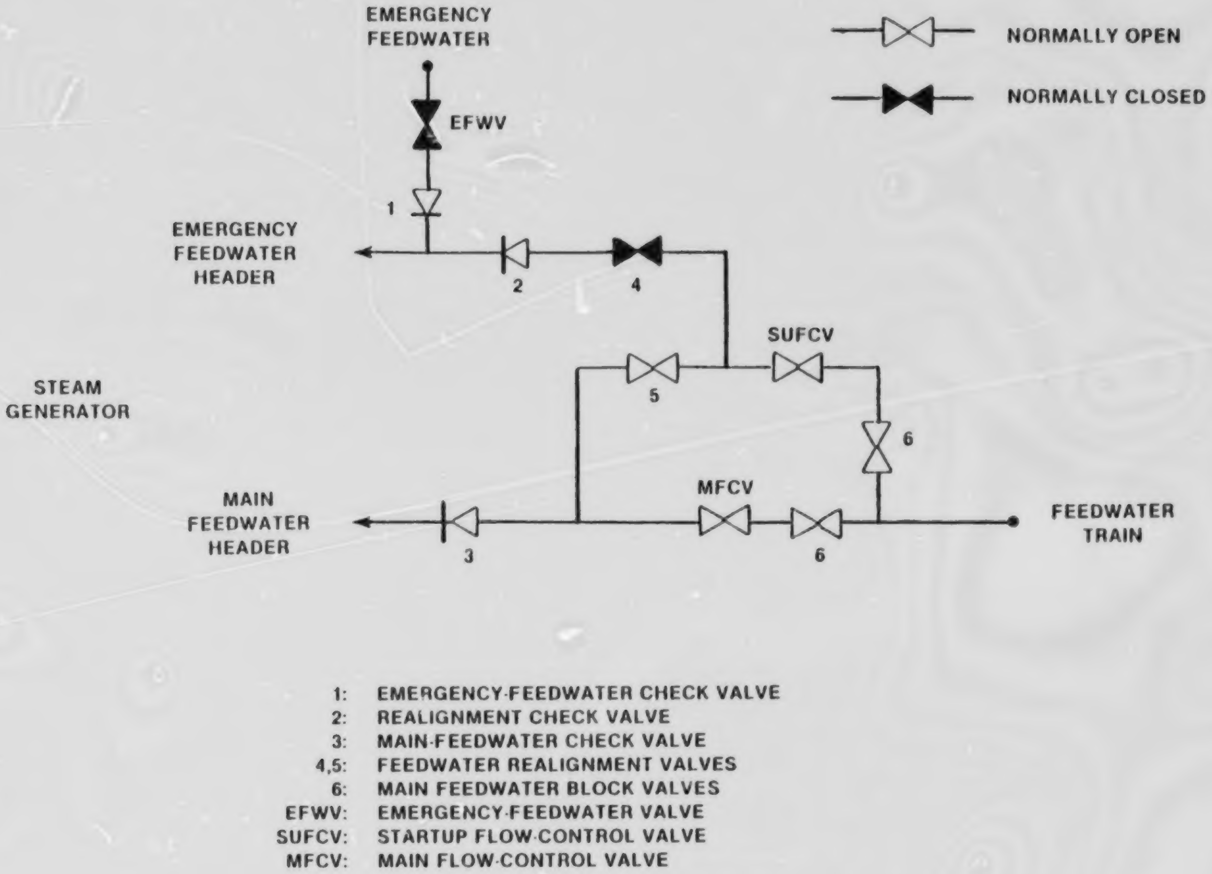
EMERGENCY FEEDWATER
EFWV
1
EMERGENCY FEEDWATER HEADER
2
4
NORMALLY OPEN
NORMALLY CLOSED
SUFCV
5
6
STEAM GENERATOR
MFCV
MAIN FEEDWATER HEADER
3
6
FEEDWATER TRAIN
1: EMERGENCY-FEEDWATER CHECK VALVE
2: REALIGNMENT CHECK VALVE
3: MAIN-FEEDWATER CHECK VALVE
4,5: FEEDWATER REALIGNMENT VALVES
6: MAIN FEEDWATER BLOCK VALVES
EFWV: EMERGENCY-FEEDWATER VALVE
SUFCV: STARTUP FLOW-CONTROL VALVE
MFCV: MAIN FLOW-CONTROL VALVE

Figure 2.7 Oconee-1 feedwater valve schematic.

the steam generator to the natural circulation set point level, or 60% of the indicated range, thus enhancing natural circulation with less overcooling of the primary.

2.2.2.2 Emergency Feedwater Train

The emergency feedwater (EFW) train consists of two motor-driven pumps (one to each steam generator) and one turbine-driven pump whose discharge is cross-connected to the discharges of both motor-driven pumps. The motor-driven pumps draw low-pressure, low-temperature subcooled condensate from the upper surge tanks and/or the hotwells. High-pressure subcooled feedwater from the combined emergency turbine- and motor-driven pumps continues to the secondary-side emergency feedwater inlet of each steam generator through independent lines containing a steam generator level control valve. The level control function is independent of the ICS. A one-way cross-connect exists for each steam generator downstream of the level control valve to allow main feedwater to enter the emergency feedwater lines (Figure 2.7). Insignificant heating of the feedwater occurs in the emergency feedwater train. Check valves throughout the emergency feedwater train prevent improper reverse flows throughout the train.

The emergency feedwater pumps are activated when the main feedwater pumps trip off or if MFW pressure drops below 750 psig. The emergency feedwater pumps continue running until shut off manually or tripped because of mechanical failure.

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2.2.2.3 Steam Generators

The configuration of the once-through steam generators is depicted in Figure 2.3. Subcooled main feedwater enters (vertically downward) the secondary side of each once-through steam generator (labeled A and B) through a spray header system, which is located approximately at the midsection of the steam generator. Steam aspirated from the steam generator shell region mixes with the feedwater as it travels through the steam generator downcomer. The feedwater is close to saturation temperature as it leaves the steam generator downcomer. Leaving the downcomer, the fluid turns vertically upward in the secondary-side (shell) region of the tube bundle where it is converted to saturated steam before reaching the level of the aspirators (slightly below the level of the main feedwater spray header). It is superheated during the rest of its passage up the shell side of the steam generator. Steam exits the steam generator by turning downward into an annulus that leads to the main steam outlets (located slightly above the main feedwater spray header). Emergency feedwater is injected radially through a spray header directly into the secondary side of the tube bundle at a level near the top of the tube bundle. The emergency feedwater spray header is intended to provide adequate cooling to assure natural circulation until the steam generator level reaches 50 % of the indicator range, corresponding to 6.1 m (20 ft) of water, at which level natural circulation will continue.

Level in the steam generator is determined from differential pressure measurements taken from taps, as indicated in Figure 2.3. The Oconee-1 plant is equipped with a main feedwater overfill protection feature. The

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protection system monitors the steam generator differential taps and trips the main feedwater pumps when either steam generator exceeds 90% of indicator range.

2.2.2.4 Main Steam Lines

The main steam lines run from the steam generators to the main steam line penetrations of the containment and into the turbine building. Eight safety/relief valves located on each main steam line limit steam line pressure by venting to the atmosphere excess main steam line mass flow that cannot be accommodated by the turbine bypass steam lines. The relief valve set points range from 7.24 MPa (1050 psig) to 7.61 MPa (1104 psig). With all 16 valves open, 114% of the design steem flow may be vented to the atmosphere.

Unlike many other plants, Oconee-1 has no main steam isolation valves. Down-stream from the safety valve taps, there are branches to several possible destinations; the condenser via the turbine bypass steam lines, the steam turbines that drive both main feedwater pumps and the turbine-driven emergency feedwater pump, the reheater/moisture separator assemblies, the condenser air ejector assembly, and the turbine stop valves. The turbine governor valves and the high-pressure turbine are downstream of the turbine stop valves.

A manually actuated atmospheric dump valve is provided on each main steam line. Because of the large time lag expected for manual operation of these valves, it is unlikely that they could contribute significantly to the progress or control of an overcooling event.

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The turbine bypass valves are used to control steam system and RCS temperature during low-power and transient conditions. There are two of these valves on each steam line, making a total of four valves. The turbine bypass valve system can vent 25% of full-power steam flow to the condensers. The control of these valves is discussed in Section 2.2.3.

2.2.3 Control Systems

The integrated control system (ICS) is a comprehensive system that controls all power generation operations of the plant. Separate safety instrumentation and controls are provided for mitigation of and recovery from transient conditions. Nevertheless, the scope of the ICS is such that it can significantly influence the course of an overcooling event. Figure 2.8 diagrams the Oconee-1 control system hierarchy. The higher-order segments such as the auto dispatch system and unit load demand generate a power demand signal that serves as a set point for the lower control systems. The integrated master system coordinates the activities of the reactor control, feedwater control, and (secondary steam) pressure control systems to compensate for upsets in any of the lower-order systems and to satisfy the demands of the higher-level systems.

The contributions of the ICS to PTS risk are twofold. First, failures in the ICS can initiate reactor trip and such overcooling events as steam generator overfeed and turbine bypass valve control failure. Therefore, the ICS can influence the frequency and, to an extent, the type of event initiators. Second, after the initiation of an overcooling event, portions of the ICS (namely, feedwater control and pressure control) can influence the response of the plant to the initiator. The structure of the ICS

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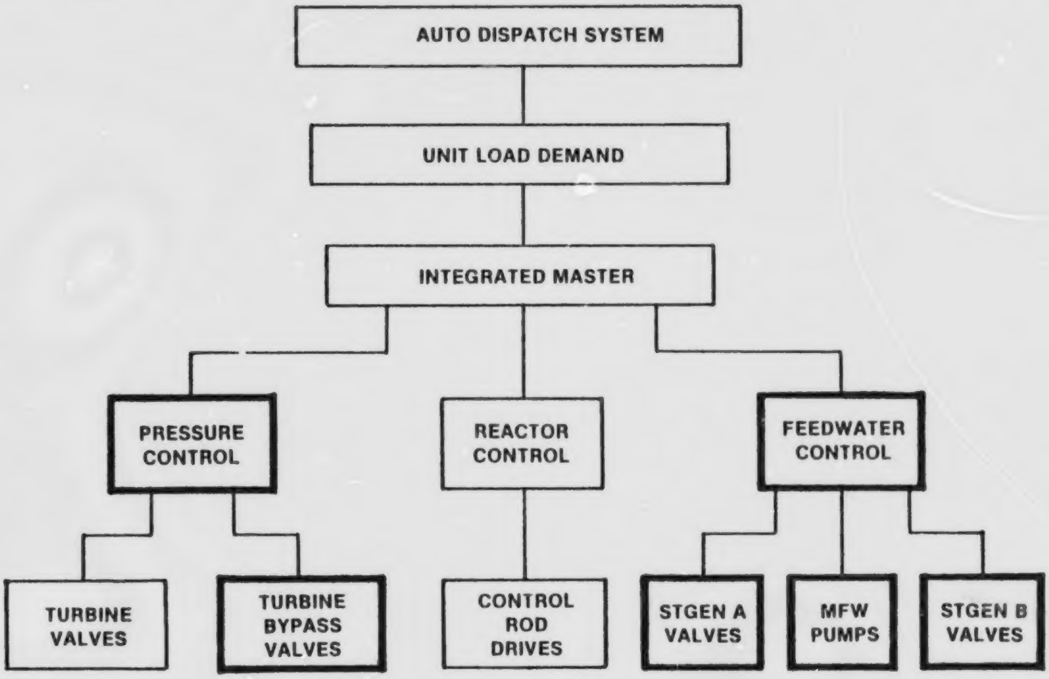
AUTO DISPATCH SYSTEM
UNIT LOAD DEMAND
INTEGRATED MASTER
PRESSURE CONTROL
REACTOR CONTROL
FEEDWATER CONTROL
TURBINE VALVES
TURBINE BYPASS VALVES
CONTROL ROD DRIVES
STGEN A VALVES
MFW PUMPS
STGEN B VALVES

Figure 2.8 Oconee-1 control systems hierarchy.

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pertaining to feedwater control and pressure control is important to the thermal-hydraulic analysis of the plant.[2.7] These systems are described in the following sections.

2.2.3.1 ICS Feedwater Control Logic

This section briefly summarizes the control algorithm implemented in the ICS for feedwater control. The discussion includes the necessary signals from the unit load demand and reactor control systems and describes the feedwater control logic concisely without detailing other ICS functions not directly related to PTS. The different elements of the ICS feedwater control system are diagrammed in Figure 2.9.

Unit Load/Reactor/Feedwater Demand Characterization

The unit load demand subsystem (ULD) establishes the demand for megawatt generation of the unit. Many conditions within the unit can result in preventing the unit from generating the requested megawatts. In this case, the unit must follow the limiting variable in order to maintain coordination of all the control variables. This mode of operation is termed "tracking," and the generated power signal is used as the indication of the limited variable. Although several conditions cause a transfer to the tracking mode, only reactor trip is pertinent in the transients to be considered. Since a turbine trip also produces an anticipatory reactor trip, this condition is included. On reactor trip, ULD tracks generated megawatts at a maximum rate of 20% full power per minute (FP/min). Since the generated megawatts will fall off much faster than 20% FP/min, the ULD can be modeled effectively with a 20% FP/min ramp initiated at

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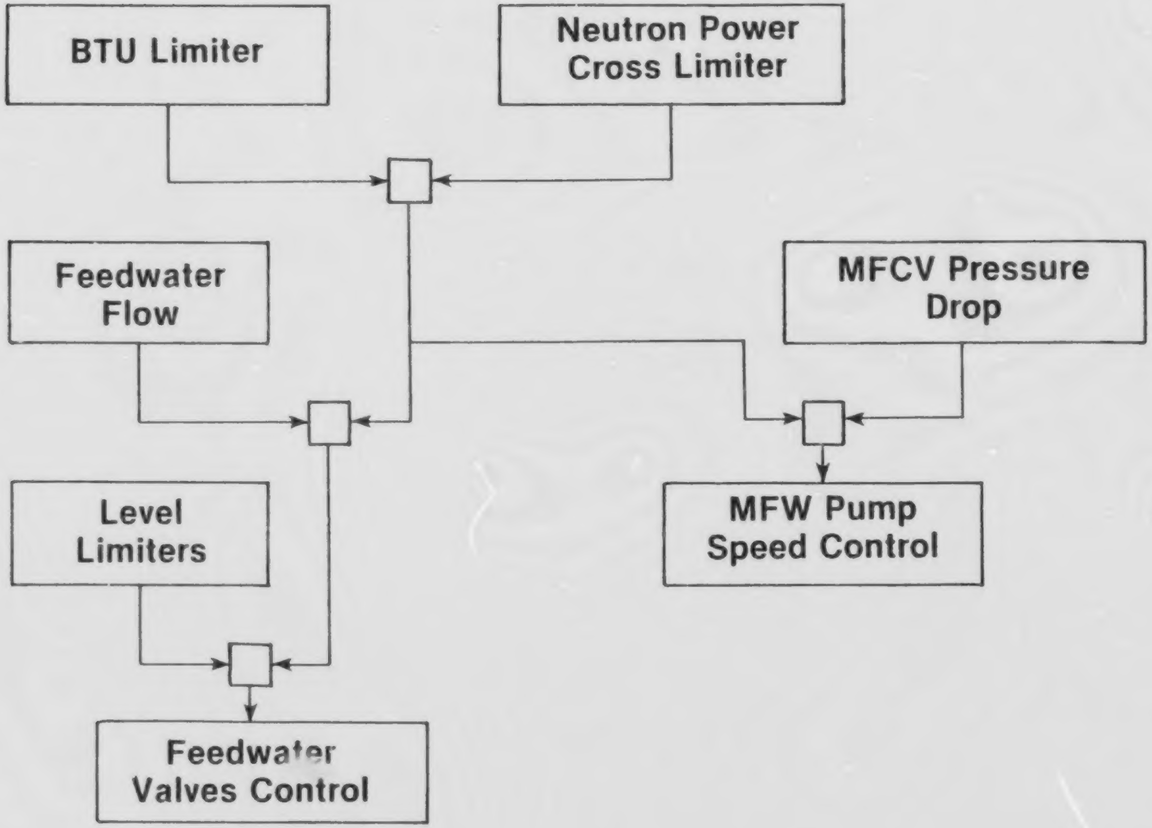
BTU Limiter
Neutron Power Cross Limiter
Feedwater Flow
MFCV Pressure Drop
Level Limiters
MFW Pump Speed Control
Feedwater Valves Control

Figure 2.9 ICS feedwater control scheme.

reactor trip. If one or more reactor coolant pumps or main feedwater pumps are tripped before ULD runback is complete, then the runback rate becomes 50% FP/min instead of 20% FP/min. Since the primary runback signal to feedwater control is the neutron cross-limit and the Btu limit, this is not considered significant. If operational transients were to be considered, the additional detail would be required.

Reactor demand is scaled from ULD to provide the required reactor power at steady state. This scaling is necessary since full-scale ranges of reactor power, feedwater flow, and generated megawatts are somewhat different. The reactor demand signal is then limited between 10% and 103%. The neutron power signal is compared with the reactor demand signal to develop the neutron power error. The resultant signal is sent to the feedwater control subsystem for cross-limiting feedwater demand.

Feedwater demand is characterized from the ULD to provide the required steam flow for a given power level at steady state. This characterization is such that the nominal feedwater flow rate is requested at full-power conditions.

Compensated Feedwater Demand

The steam generator demand is characterized to represent the expected feedwater temperature. This signal is compared to the actual feedwater temperature to produce a feedwater temperature error signal that compensates the feedwater demand.

The neutron error signal from the reactor control subsystem can also modify the total feedwater demand. If the neutron error becomes greater than

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± 5%, the total feedwater demand will be modified to maintain the proper ratio between feedwater flow and reactor power. If this signal is greater than ± 5%, the unit is said to be operating under "cross limits." Note that after a reactor trip, it is the cross-limit signal that initiates rapid runback of feedwater flow since the ULD ramp rate is merely 20% FP/min.

Feedwater Loop A and B Demand

Separate feedwater flow demands are developed for loop A and B. In the ICS, these demands are characterized to meet the requirements of total feedwater demand and to maintain a set point temperature differential (normally zero) between the two reactor cold legs. Since this ratio controller output is blocked when either steam generator is on a level limit or during a transient, it is not important. Consequently, the total feedwater demand signal can be sent unmodified to each loop in the model used for PTS studies.

BTU Limited Feedwater (FW) Demand

The outlet steam pressure of each steam generator is characterized to produce a signal equivalent to the FW flow limit for that pressure. Likewise, the FW temperature, the selected reactor outlet temperature, and primary loop flow are conditioned to produce equivalent FW flow limit signals. These signals are then used to develop a Btu availability limit signal, which limits the feedwater demand for the steam generator to prevent a reduction in steam temperature. The Btu signal is then auctioneered against the loop demand signal. The limiting signal (lower

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FW demand) is used to develop the loop demand for MFW pump control and in the development of the FW flow error signal.

Feedwater Flow Error

The loop FW flow signal is compared with the FW flow demand to produce the loop FW flow error. The total FW flow in the loop is obtained from the sum of the startup and main feedwater flows. The measured flow is fed through a lag unit to produce a more stable control signal.

High and Low Steam Generator Downcomer Level Limits

The FW flow error signal is compared with high- and low-level error signals. Thus, if the FW flow error demands an increase in FW flow and the downcomer level is at the maximum allowable, the level error signal will assume control to prevent flooding of the aspirating steam ports. Similarly, if the FW flow error tries to reduce the FW flow to a point where the downcomer level drops below the minimum allowable value, the low-level error signal will take control.

Feedwater Valve Control

The level-limited FW demand error is used for control of the main and startup FW valves. Proportional plus integral (P + I) action is applied to the error signal from two different controllers. During level control at the low-level limit, one of the controllers is used while the output of the other controller is blocked. When not on low-level control, the level controller is blocked and the other controller provides the control

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signal to the valves. Smooth transfer is provided when switching from one control signal to the other.

Feedwater Pump Control

Loop A and B FW valve differential pressure signals are auctioneered to select the signal representing the smaller differential pressure. The selected signal is then compared with the minimum differential pressure set point to produce an error signal. This error signal is modified with a gaining factor, which increases as the error increases. Proportional and integral action is applied to this error signal. The resultant signal is used to modify the total pump demand signal.

The modified FW demand signals from loop A and loop B are summed to produce a total FW demand signal used as the demand signal for the FW pumps. On a load change, this signal will anticipate a change in differential pressure and cause the pump speed to change, thus minimizing the error.

Feedwater Control System Set Points

The set points used in modeling the Oconee feedwater control system are given below.

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Limit or Set Points	Value	Remarks
Expected FW temperature	Function developed by ICS	
Btu limits	Function developed by ICS	
High level limit	8.14 m (344 in.) H_2O	
Low level limit	0.64 m (25 in.) H_2O	RC pump(s) operating
Low level limit	6.1 m (240 in.) H_2O	RC pump(s) tripped
FW valve ΔP	0.24 MPa (35 PSID)	

2.2.3.2 Description of Turbine Bypass Valve Control

Turbine control is a portion of the integrated master subsystem of the ICS. Since the simulation models do not include the turbine, modeling the turbine control valve is not necessary. Thus, the turbine header pressure is a boundary condition for the ICS model.

Turbine bypass control provides high-pressure relief if the turbine header pressure exceeds its set point by 0.34 MPa (50 psi) under normal conditions or by 0.86 MPa (125 psi) if the reactor is tripped. The turbine header pressure error is applied through proportional plus integral control.

Independent pressure relief is provided if the outlet pressure of either steam generator exceeds its set point. This relief is direct, proportional control action. The turbine bypass control system selects whichever control signal requires more bypass steam flow around the turbine.

On loss of condenser vacuum, the turbine bypass valves are shut to avoid sending further steam to the condenser.

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2.2.3.3 Description of Emergency Feedwater Valve Control Logic

Emergency feedwater control systems manipulate a control valve to maintain level in the steam generator. The startup level is maintained if the reactor coolant pumps are operating, and a 6.1-m (20-ft) level is maintained if they are tripped. This is to encourage natural circulation during conditions when the reactor coolant pumps are tripped. The actual steam generator level (as determined for MFW control) is compared with the level set point, and EFW flow to the steam generator is adjusted accordingly.

There is no automatic isolation of feedwater to a steam generator with a steam line break or similar depressurization event.

2.2.4 Support Systems

Support systems supply power or cooling to many of the major systems and components described in Sections 2.2.1 to 2.2.3. These support systems were considered because of the potential of single failures of support systems to lead to overcooling transients in nuclear power plants. The principal support systems in the Oconee-1 plant are

Cooling water systems (CWS),
Instrument air systems (IAS), and
Electrical and instrumentation power systems (EPS).

Each of these systems has redundant busses or trains or alternate backup systems. The detailed description of these systems is beyond the scope of this section. An interaction evaluation of the support systems was

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performed to determine which failures would initiate or influence PTS sequences.[2.8] Six "systems" were identified as main candidates, namely:

(1) PORV

(2) HPI/reactor coolant makeup and purification (MU&P)

(3) Turbine bypass valves (TBV)

(4) Turbine trip

(5) Main feedwater

(6) Emergency feedwater

The PORV (and its controls) was selected since its failure in the open position is a small LOCA, one of the PTS initiating transients of concern. The HPI/MU&P system is required to operate to repressurize the RCS in almost all PTS transients considered. In addition, the HPI/MU&P provides RC pump seal injection, which is important to the prevention of RC pump seal failure. The turbine trip function and the TBV were selected since untripped or open failures could result in steam generator depressurization and RCS cooldown. The main or emergency feedwater systems' function is required in all non-LOCA PTS transients to cause the RCS overcooling condition.

The results of the interaction study are summarized in Table 2.2. The PORV and its control circuitry do not depend on the the CWS directly or on the IAS. The valve will be closed in the event of associated EPS failures.

The HPI/MU&P systems require cooling water, but a single failure is overcome by manual changeover from low-pressure service water (LPSW) to high-pressure service water (HPSW). Instrument air failures will cause the

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Table 2.2 Failure modes of systems in response to support systems single failures[*]

System	CWS Failure	IAS Failure	EPS Failure
PORV	Operable	Operable	Closed[**]
HPI/MU P			
o HPI	Operable/off	Operable	Operable/off
o Makeup control valve	Operable	Closed	Manually/ operable
o RCP Seal Inj. control valve	Operable	Open	Throttled
TBV	Operable/closed	Closed	Closed[**]/ manually operable
Turbine trip	Tripped	Tripped	Tripped
Main feedwater	Tripped	As is	Throttled
Emergency feedwater	Operable/off	Operable	Operable

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[*] Letter from R.L. Gill, Jr. (Duke Power) to R.C. Kryter (ORNL), October 19, 1982.

[**] Telecon with D.W. Henneke (Duke Power), November 11, 1982. Source: Reference 2.8.

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makeup control valve to fail closed and the RCP seal injection valve to fail open. EPS failure will, depending on which supply is lost, throttle the valves and affect HPI motor control (i.e., start or stop motors already running).

The turbine bypass valves will fail closed or be manually operable for any single support system failure. Multiple failures causing erroneous signals from the ICS may cause the valves to fail 50% open.

A disabling failure of any of these support systems will result in turbine trip. Main feedwater will trip on loss of cooling to the feedwater pumps. Loss of IAS will freeze the feedwater regulating valves in position. The valves will throttle on EPS failure-related loss of control signal.

Because of backup electrical and air systems, the emergency feedwater systems will initially function. Loss of backup pressurized-nitrogen systems would cause the EFW control valves to fail open. Loss of cooling could cause eventual failure or trip of the EFW pumps.

2.3 Operations

The Oconee-1 plant systems and operating procedures are intended to provide sufficient cooling to prevent core damage for sequences up to the design basis accidents. Likewise, the design and operations also seek to limit RCS cooldown rates to 55.56 K (100°F)/h, the design cooldown rate for reactor vessel protection. RCS temperature and pressure response to overcooling events are influenced by three main mechanisms:

(1) Automatic system responses,

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(2) Operator intervention, and

(3) Passive features.

Automatic responses include reactor and turbine trips, feedwater runback by ICS, turbine bypass valve and PORV operation, HPI initiation, EFW initiation and control, and other assorted automatic trips. Operator interventions include tripping and starting of reactor coolant pumps, throttling of HPI, and manual control or isolation of selected systems. Such actions are governed by written procedures. Passive features include the safety relief valves for RCS and steam system pressure control, reactor vessel internal vent valves, and design-related phenomena such as RCS natural circulation. The normal interplay of these actors for overcooling sequences is discussed below.

2.3.1 Reactor Trip

The normal reactor trip sequence represents the baseline against which other more severe transients may be compared. Prior to the reactor trip, the core power level is steady, the reactor coolant pumps are operating, and the ICS is maintaining a constant steam flow corresponding to system power output. The normal primary (RCS) flow paths are diagrammed in Figure 2.10. The initiating event is assumed to be external to the nuclear and steam-generating systems. A turbine trip satisfies this assumption. The turbine trip simultaneously triggers the reactor trip and begins closure of the turbine stop valves in each main steam line. The ICS commences runback of main feedwater by closing the main flow control valves (Figure 2.7), throttling the startup flow control valves, and reducing main feedwater pump speed. The ICS action is to track reactor power following the trip

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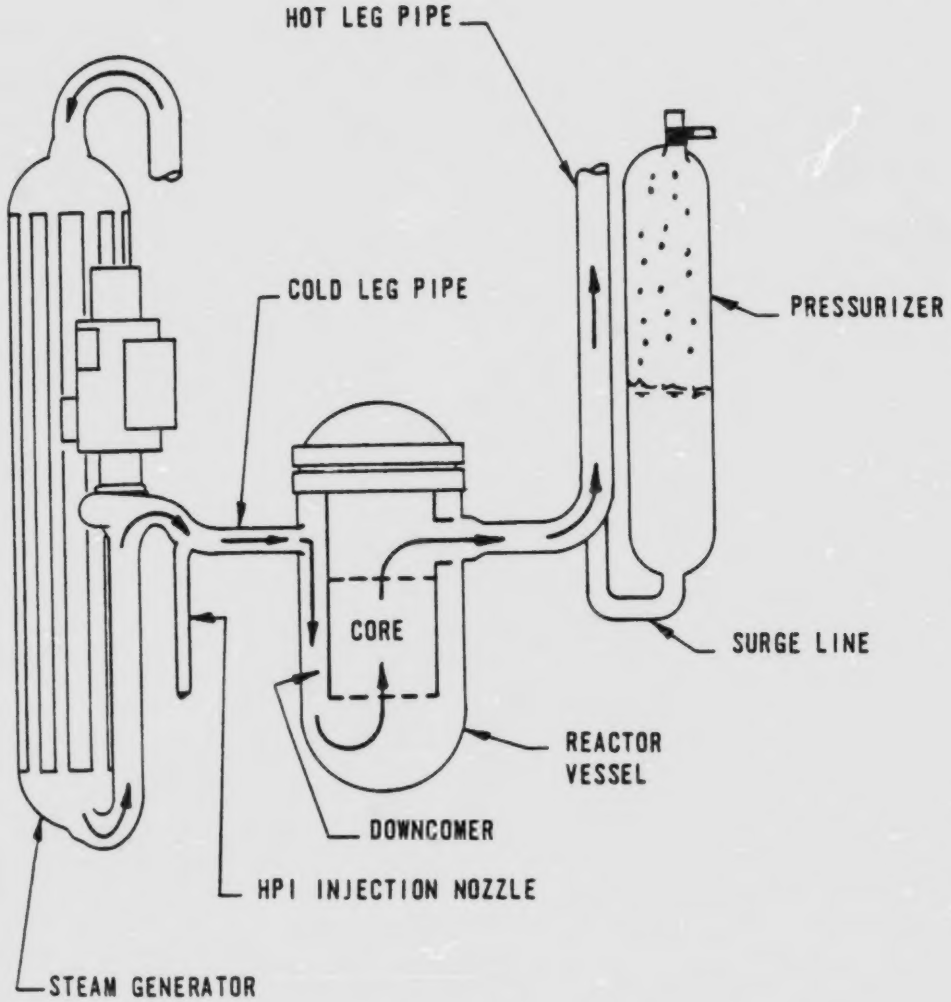
HOT LEG PIPE
COLD LEG PIPE
PRESSURIZER
CORE
SURGE LINE
REACTOR VESSEL
DOWNCOMER
HPI INJECTION NOZZLE
STEAM GENERATOR

Figure 2.10 RCS flow during normal operation.

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and maintain steam generator water inventory without overfeeding the steam generators.

After the turbine stop valves close, the pressure in each steam line rises and the ICS opens the turbine bypass valves to relieve the pressure and commence cooldown of the plant. Several of the steam safety relief valves may also open momentarily after turbine stop valve closure. As the RCS heat inventory and reactor power continue to decrease, the steam generation rate falls and the ICS modulates the TBVs to maintain constant steam generator pressure 6.97 MPa (1010 psig) and, indirectly, constant RCS temperature. Therefore, the plant is brought to a steady condition with the average RCS temperature dropping to about 560 K (549°F) from 577 K (579°F). Operator-implemented [illegible]own procedures are then used to bring the plant to shutdown condition.

The RCS also undergoes a pressure response related to its temperature response. Following reactor trip, heat transfer to the secondary side causes reduction in coolant temperature and specific volume. This coolant shrinkage results in liquid flow out of the pressurizer, which in turn leads to steam expansion in the pressurizer dome and a drop in RCS pressure. The pressurizer design volume is small relative to the rest of the RCS so that coolant shrinkage from mild transients can lead to loss of pressurizer water level indication and tripping of the pressurizer heaters. To maintain pressurizer water level, the operator may manually start a second HPI pump to complement the normal makeup flow. (This operator action was not modeled in the transient analyses for this study.) The makeup flow stabilizes the RCS pressure and will cause repressurization

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once the system cooldown rate decreases. If the operator does not throttle the HPI/makeup system at some lower pressure, the RCS will reach the PORV set point pressure 16.9 MPa (2450 psig) where it will be regulated by the action of the PORV. Written procedure specifies that the operator throttle the makeup system on attainment of 27.8 K (50°F) subcooling in the RCS.

The flow regime in the RCS is unaffected in this transient since the reactor coolant pumps are not tripped.

2.3.2 LOCA Events

Loss-of-coolant accidents (LOCA) have the potential to generate stagnated flow regimes that can contribute to localized rapid cooldown of the reactor vessel. Depending on break size, the RCS may remain at high pressure (small breaks) or drop to low pressures (intermediate and large breaks). The location of the break can also influence system flows and cooldown rates. The general system responses to this type of event are described below.

Prior to initiation of the break, the normal circulation of the coolant is as shown in Figure 2.10. After the break opens, the loss of coolant causes a drop in pressure, which will cause reactor trip on P-T relationship or low-pressure 12.41 MPa (1800 psig) and initiate HPI at 10.34 MPa (1500 psig). Operating procedures require the operator to trip all four reactor coolant pumps following actuation of HPI. For very small breaks, the HPI system capacity is greater than the break flow, giving the operator control over system pressure and allowing orderly cooldown. For small breaks where the HPI capacity matches the break flow, the pressure will seek the value dictated by the HPI pump head-capacity curves (Figure 2.5). Such transients are characterized by a semisteady flow and pressure conditions with temperature slowly declining as decay heat decreases. Beyond a certain break size, the HPI cannot maintain pressure, so RCS pressure may drop to 4.14 MPa (600 psig) where CFT flow begins or to 1.38 MPa (200 psig) where LPI flow begins.

With the reactor coolant pumps tripped, continued heat transfer from the primary to the secondary through the steam generators will induce coolant density differences around the cooling loops and cause natural circulation flow. This condition will continue until either steam voids collect in the candy cane(s) interrupting flow or the RCS becomes cooler than the steam generators, setting up opposing density gradients that result in a zero net driving force for natural circulation flow. Despite the stagnation of loop flows, the differences in downcomer and core region coolant density are sufficient to yield circulation through the vent valves (see Figure 2.2 and Section 2.2.1). There is also a net flow towards the location of the break. Figure 2.11 depicts the flow regime for a break at the top of the pressurizer. The vent valves are designed to promote continued circulation of coolant through the core region as well as provide warm water to mix with HPI flows entering the downcomer region.

2.3.3 Secondary-Side Overcooling Events

Secondary-side overcooling events include steam line breaks and feedwater overfeed events. Steam line breaks include stuck-open valves as well as pipe breaks. The initial response of the RCS to a significant overfeed or steam line break is similar to the response to a small-break LOCA; that is, RCS pressure drops below the HPI initiation pressure and temperature

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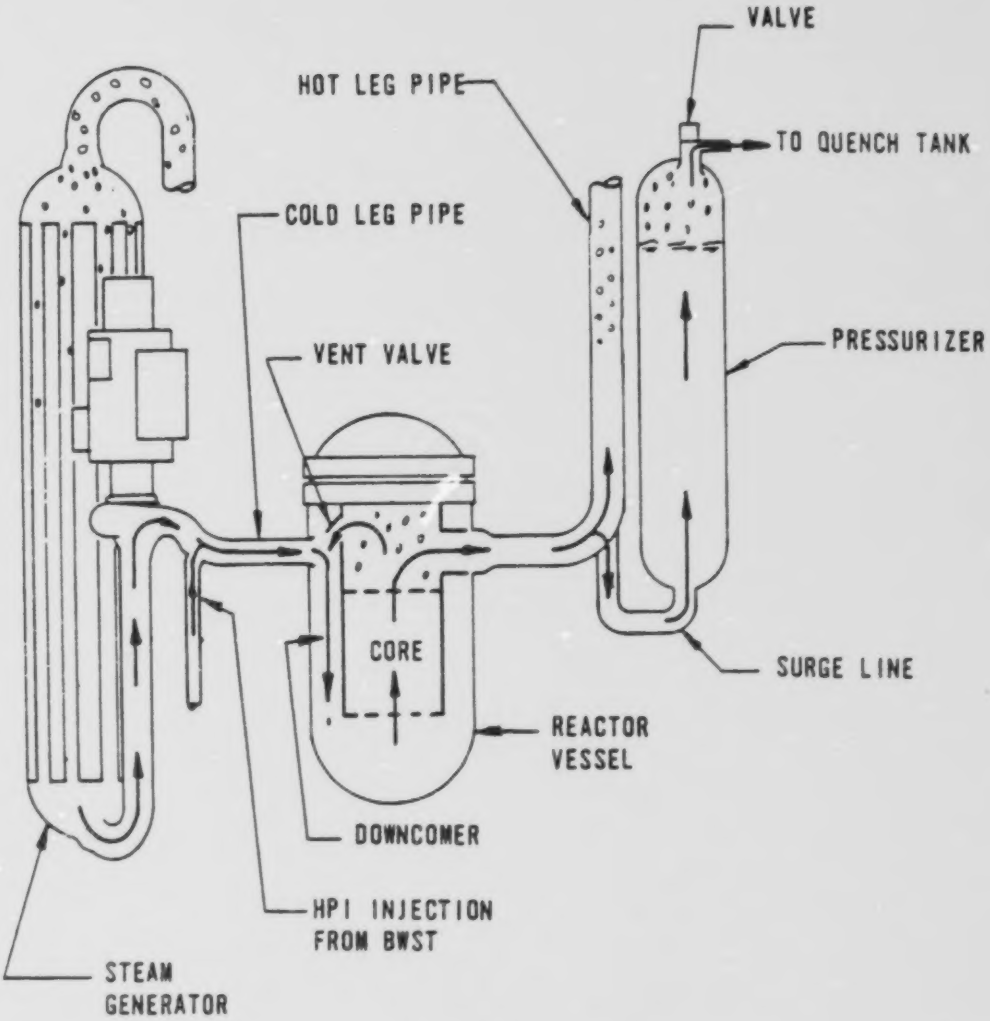
VALVE
HOT LEG PIPE
TO QUENCH TANK
COLD LEG PIPE
PRESSURIZER
VENT VALVE
CORE
SURGE LINE
REACTOR VESSEL
DOWNCOMER
HPI INJECTION FROM BWST
STEAM GENERATOR

Figure 2.11 RCS flows during SBLOCA at the top of the pressurizer.

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declines rapidly. For this reason, the operator will handle the sequence as a LOCA (i.e., trip the RCPs) until continued monitoring of the RCS and steam generators allows elimination of LOCA concerns and confirmation of an overfeed or steam line break.

Since the Oconee-1 plant is not equipped with main steam isolation valves or main feedwater and EFW isolation systems and supporting logic systems, feedwater isolation and TBV isolation, are accomplished by manual control.

The flow regime behavior is similar to that of the small-break LOCA described above. The reactor coolant pumps are tripped following actuation of HPI. Natural circulation is initially established in both cooling loops. This circulation will be for "symmetric" conditions such as overfeeds to both steam generators or breaks affecting both steam lines. If only one steam generator is subject to overcooling, natural circulation flows will be nonsymmetric. The loop containing the intact steam generator may stagnate if the RCS is cooled below the intact steam generator temperature, causing cancelation of flow-driving potential for the loop. For severe cooldown and primary depressurization events, voiding of the candy cane may physically block flow of liquids. With the loss of loop flow, the vent valves will allow continued core region to downcomer flow as in the LOCA case, although flow rates will vary.

If the affected loop secondary dries out or is isolated, the system will reheat until natural circulation is reestablished in the intact loop. The intact loop then can be used for system heat removal during final shutdown.

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REFERENCES

2.1. Duke Power Company, "Oconee Nuclear Station Final Safety Analysis Report," Chapters 4, 6, 7, 8, 9, 10.

2.2. Babcock & Wilcox, "Steam, Its Generation and Use," The Babcock & Wilcox Company, New York, 1978. Chapters 21, 23, 35.

2.3. Duke Power Company, "Reactor Vessel Pressurized Thermal Shock Evaluation," DPC-RS-1001. January 1982.

2.4. J. Ireland, et. al., "TRAC Analyses of Severe Overcooling Transients for the Oconee-1 PWR," LA-UR-83-3182 (no date). pp. 33, 34.

2.5. C.D. Fletcher, et. al., "RELAP5 Thermal Hydraulic Analysis of Pressurized Thermal Shock Sequences for the Oconee-1 Pressurized Water Reactor," EGG-NSMD-6343, July 1983. p. 97.

2.6. "A Pressurized Thermal Shock Scenario Compatible RELAP5 Thermal-Hydraulic Model of the Oconee-1 Steam and Power Conversion System Main/Emergency Feedwater Trains," SAI #1-245-08-388-01, prepared for Instrumentation and Controls Division, ORNL by Science Applications, Incorporated, Oak Ridge, Tennessee, December 1982.

2.7. "Overview of Integrated Control System Modeling for the Pressurized Thermal Shock Study," prepared for ORNL by Science Applications, Incorporated, Oak Ridge, Tennessee, October 13, 1982.

2.8. A.F. McBride, letter to J.D. White, ORNL, "Oconee Interaction Analysis - Incorporation of Duke Power Comments," May 10, 1983.

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3.0 EVENT SEQUENCE ANALYSIS

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R. Olson, Science Applications, Inc.

3.1 Introduction

The purpose of the event sequence development and analysis in this project was to document the event sequences that could involve significant potential for pressurized thermal shock (PTS), to select sequences with a range of severities for detailed thermal-hydraulic analysis, and to estimate the probability of occurrence of all sequences.

A straightforward application of event tree methods to the PTS problem would result in an entirely unmanageable number of sequences, far more than would result if core damage were being investigated, and far more than could be studied with thermal-hydraulic analysis at any reasonable cost. The event tree task thus used a modular approach. System state trees were developed for the major plant systems independently of the transient initiating events. These event trees were made to be applicable to the various initiating events by incorporating "conditioning events," that contain the proper system configuration (i.e., pressure, temperature) for each initiating event. Human error or actions were coded into "operator action trees" separately from the system state trees, and these trees were used to modify the outcomes of the system state trees for each initiating event, as appropriate.

The methodology is described in Section 3.2, the construction of the system state trees is discussed in Section 3.3, and the quantification of the

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trees is discussed in Section 3.4. The way in which the transients were categorized for consideration of operator action is shown in Section 3.5, and the construction of the initiator-specific system state trees is described in Section 3.6. The construction and quantification of the operator action trees are described in Section 3.7.

3.2 Methodology

The representation of PTS transients using the "standard" event tree formalism is extremely difficult because of the complexity inherent in the PTS phenomena. The principal reason for this complexity is the dependence of the consequences of PTS transients (the failure/non failure of the pressure vessel) on the temperature history of the near-wall primary coolant as well as the pressure. Thus, the timing of various system actions during a transient can have a significant impact on the consequences of the transient. A second complexity is related to the fact that for PTS transients, the consequences may not be monotonic functions of the principal transient parameters. Thus, intermediate values of the parameters (e.g., valve positions) may result in more severe consequences then either of the two extreme positions (i.e., fully open or fully closed). Additionally, PTS transients are usually characterized by strong interactions among thermal-hydraulic parameters, and these interactions must be properly accounted for in any tree-type representation. Finally, there is the question of operator actions and, more specifically, the timing of operator actions. For some PTS-related transients, mitigating actions by the operator are the only mechanism for preventing challenges to the pressure vessel. However, successful mitigative action is dependent

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on the time available for such action, which is, of course, dependent on the specific transient.

Rather than try to incorporate all of the above effects into standard event tree formalism, an alternate methodology was developed and used for this. Two sets of trees were developed to represent PTS-related transients, a group of system state trees and a set of operator action trees. The system state trees were designed to permit the ready identification of the possible system configurations and to assist in the specification of the transient sequences to be subjected to detailed thermal-hydraulic moduling and calculation. The rationale for separate operator action trees (as opposed to incorporating operator actions into an event tree) is two fold: (1) the need to consider various timing options for specified operator actions, and (2) the recognition that operator action under PTS conditions may require cognitive processes in addition to purely mechanistic actions. The separation of the operator actions analysis from the system response analysis facilitates both objectives. It should be emphasized that the use of separate trees for the system and operator responses does not imply that they are independent conceptually. They should be viewed as parallel trees (in time) with a high level of interaction.

The system state trees were developed on a functional basis. Since PTS transients typically involve a significant number of systems and subsystems, it was felt that a functional level approach would serve to clarify both the reasons for the modeling as well as the interrelationship between the systems. The four functional groupings used in this study were:

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1. Secondary pressure control (steam lines, turbine stop valves, turbine bypass valves, and steam relief valves),

2. Steam generator feedwater supply (main feedwater and emergency feedwater systems),

3. Primary loop coolant flow (reactor coolant pumps, high-pressure injection, core flood tanks, low-pressure injection, and vent valves), and

4. Primary loop pressure control (PORV, safety relief valves, pressurizer sprays and heaters).

These four functional groupings were found to be inclusive enough to encompass the transients that result in low fluid temperatures coupled with (or followed by) high primary system pressure.

To facilitate the initial representation of PTS-related transients on the system state trees, two distinct types of events are considered: system response events, and conditioning events. The majority of the events represent system responses to the transient in question. In many cases the system response events include representation beyond the binary success and failure branches found on most standard event trees. Such a representation was felt to be necessary to distinguish between system states such as "one of two valves fails" and "two of two valves fail." The conditioning events represent thermal-hydraulic states and serve a dual purpose; they restrict the number of potential end-states for a given system state tree, and they permit the coupling between the various functional system state trees (due to the thermal-hydraulic interactions)

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to be represented. An example of a conditioning event is the turbine stop valve status (open or closed) dependent on the plant status (at power or at hot standby). The conditioning event terminology is used since subsequent system responses are considered conditional on the thermal-hydraulic parameters that comprise the conditioning event description.

Initially, the system state trees are considered to be time-independent. From such a perspective, the end-states can be viewed as an asymptotic distribution of possible system configuration. It must be noted that such an interpretation implicitly contains three assumptions:

1. No operator action to mitigate (or exacerbate) the transient,
2. Infinite capacities for reservoirs involved in the transient (e.g., the borated water storage tank), and
3. No thermal-hydraulic limitations to the proper functioning of operating systems (e.g., the RC pumps do not cavitate under low pressure or two-phase flow conditions).

In spite of the unrealistic nature of these assumptions, the system state trees represent an extremely useful and instructive initial categorization of the extent of PTS challenge inherent in the systems. In particular, the subsequent interpretation of the trees as time-dependent, coupled with thermal-hydraulic analysis, permits the identification of inherent system limitations that serve to mitigate the severity of PTS transients. Moreover, analysis of a few of the asymptotic cases from both a thermal-hydraulic and fracture mechanics perspective allowed the estimation of reference time frames within which operator action is effective. This

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information is necessary to quantify the operator action trees for each transient because the probability of operator error is related to the time available for the action. The use of the asymptotic end-state perspective also permits a unique interpretation of the operator actions. The actions of the reactor operator(s) can be viewed as "switching" a transient from "on track" to an alternate asymptotic configuration. And, for most of the envisioned PTS transients, the end-state ultimately switched to does not require the assumptions listed above.

In general, an operator action tree must be constructed for each unique asymptotic end-state. The operator action trees are based on the premise that the transient is proceeding towards the specified asymptotic end-state, and they detail the significant action/non action available to the operator. In reality, some operator actions, particularly those involving long time displays, may be precluded by the failure to hold of one of the other two thermal-hydraulic assumptions. However, the initial operator action trees were constructed without regard to this effect.

The overall system state tree/operator action tree methodology can be summarized as a seven-step procedure:

1. The identification of the relevant conditioning events for each of the four functional groupings (secondary pressure control, feedwater, primary coolant flow, and primary pressure control);

2. The construction of generic system state trees for each functional group that incorporate the conditioning events and their appropriate system/subsystem responses;

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3. The merging of the system state trees for each of the four functional groupings into a single system state tree for a specific transient initiator;

4. The quantification of the branch probabilities based on the appropriate conditioning events;

5. The pruning of the overall system state tree based on frequency (this study utilized 1.0E-06 as a nominal cutoff), physical impossibility, and engineering judgment regarding the conditioning event;

6. The construction of an operator action tree for the remaining significant end-states;

7. The quantification of each operator action tree to determine the overall frequency of the various end-states including the effects of operator actions.

3.3 Construction of the Oconee-1 System State Trees

3.3.1 Secondary Pressure Control System State Trees

Since an excessive steam demand can result in a potential over cooling transient, the secondary-side pressure-control system is a requisite part of any PTS analysis. The mechanisms for control of the secondary-side pressure for the Oconee-1 plant include the turbine stop valves, the turbine governing valves, the secondary steam relief valves, and the turbine bypass system. The Oconee-1 plant has four turbine stop valves (TSVs) (two on each steam line) linked to a common header upstream of the four governing

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valves. Each steam line has four pairs of safety valves which open sequentially. Finally, two pairs of turbine bypass valves (one pair per steam line) permit the dumping of steam directly to the condenser and serve as a pressure-control mechanism for each steam line.

In constructing the secondary-side system state trees, it is necessary to develop the state tree for each steam line separately so that possible interactions can be accommodated. For the purposes of this discussion, the response of steam line B is assumed to be conditioned on the system state of steam line A. Since the Oconee-1 plant is symmetrical in this regard, the corollary description is straightforward. Additionally, in constructing the system state tree for steam line A, an attempt has been made to order the events to maintain chronological order where possible.

The status of the turbine stop valves and their response to a control signal are the most important considerations in describing the response of the secondary pressure-control system. For most areas of PTS concern (i.e., steam line breaks upstream of the TSV), the relevant question is whether the TSVs on steam line A close in response to a turbine trip signal, thereby isolating steam line A from both the turbine and steam line B, preventing a blowdown of both steam generators. To model this response, a binary event branch is assumed, with the success branch defined as the closure of both TSVs. Additionally, this event branch point also serves as a convenient means for treating the status of the TSVs where it is specified by the initial conditions (e.g., closed at hot standby). This knowledge is incorporated into the quantification process by setting the success branch probability equal to 1. Finally, transients can be

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postulated wherein the relevant PTS question is not whether steam line A is isolated from steam line B but whether either one or both are isolated from the turbine itself. For this case, the relevant failure branch is interpreted as the failure of at least one TSV to close on demand and the failure of at least one turbine governing valve to close on a turbine trip. However, unless specifically noted in the discussion, the initial definition of this event will be used.

If the turbine stop valves on steam line A successfully close (or are closed), the response of the remaining pressure control subsystems becomes relevant to the PTS problem. It is assumed that if the TSVs do not meet the success criterion, there will be no demand for any additional secondary pressure relief.

Given success of the TSV branch, a conditioning event based on the maximum steam line pressure occurring in steam line A is introduced. This conditioning event is used as a demand switch to identify which of the remaining two secondary pressure control systems are demanded, the safety relief valves (SRVs) or the turbine bypass valves (TBVs). Three relevant states are employed: State 0 - maximum steam line pressure greater than 1050 psi (both SRVs and TBVs demanded); State 1 - pressure between 935 and 1050 psi (only TBVs) demanded); and State 2 - maximum pressure is less than 935 psi (neither demanded). Generally, only one branch is applicable to a given transient, and the appropriate branch is determined from thermal-hydraulic analyses. However, as will be apparent in the discussion of the initiator-specific event trees, using such conditioning event switches greatly facilitates the construction of the initiator-specific event trees.

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The operability of the turbine bypass system is the next event considered in developing the secondary pressure control system state tree. Given a demand signal, three TBV response states are defined. For the first (State 0), the TBVs on steam line A function as designed to control the secondary pressure, opening on high pressure and modulating to maintain desired pressure. The first failure branch (State 1) is defined as at least one TBV's failure to modulate back from its maximum open position. For this state the possibility of the stuck-open TBV's reclosing at some point later in the transient is neglected. State 2 is defined as a failure of both TBVs to open on demand.

The next response of concern to the PTS analysis is whether, given that at least one SRV is demanded, the valve opens. For this analysis, it is assumed that if the SRVs are demanded, they will open as designed, hence a specific branch relative to the opening of the SRVs is not employed. The subsidary question of whether the SRVs, having opened as designed reseat when the steam line pressure is reduced, is represented by a separate event, which can be integrated as an induced steam line break. For the purposes of this analysis, the success branch is defined as all SRVs, which have opened on demand, reseating at pressures greater than 1000 psi. The failure branch is defined as at least one SRV's failure to reseat at a pressure > 1000 psi. Although the possibility exists that the malfunctioning valve may reseat at a lower pressure than the design value, it is assumed that the failure is in effect for the length of the transient.

Figure 3.1 depicts the system state tree that results from combining the four secondary-side pressure-control events outlined above. Additionally,

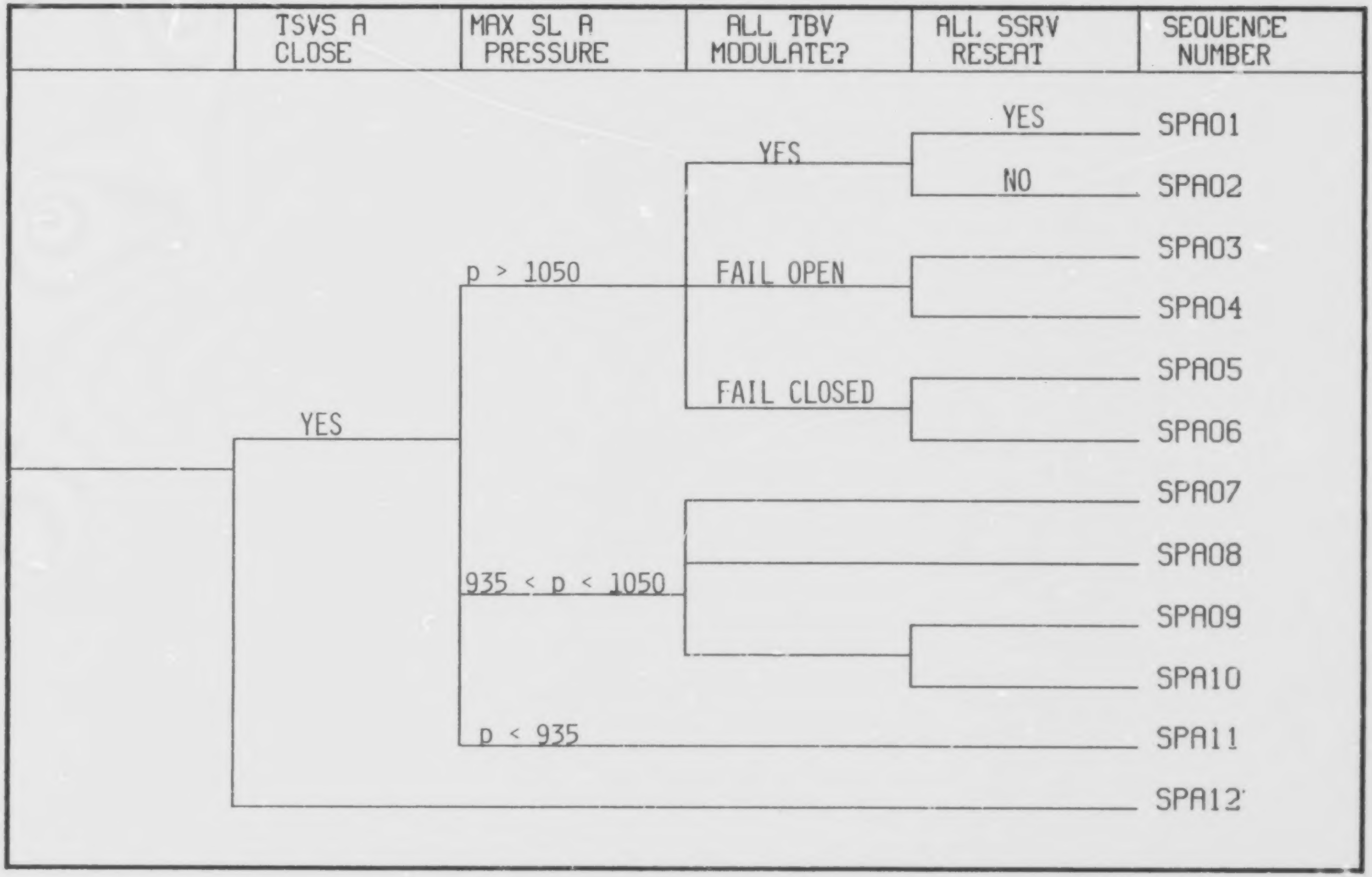
TSVS A CLOSE
MAX SL A PRESSURE
ALL TBV MODULATE?
ALL SSRV RESEAT
SEQUENCE NUMBER
YES
p > 1050
YES
YES
NO
FAIL OPEN
FAIL CLOSED
935 < p < 1050
p < 935
SPA01
SPA02
SPA03
SPA04
SPA05
SPA06
SPA07
SPA08
SPA09
SPA10
SPA11
SPA12

Figure 3.1 Secondary pressure control (SL A) system state tree; initiator: unspecified.

for convenience, Table 3.1 presents an abbreviated version of the success/failure criteria for each branch.

The system state tree for steam line B is constructed in an analogous fashion to that of A with one notable exception, that of the conditional probability of failure of the pressure-control components of steam line B given the failure of the same or related components of steam line A. Structurally, the system state tree for steam line B is identical to that for steam line A. However, as will be discussed in the following section, the assumption of a degree of commonality between components actually results in multiple system state trees for B conditioned on the system state of steam line A. At least five distinct categories of end-states can be distinguished: nominal operation (sequences SPA01, 05, 07, 09, 11), open TBVs (SPA 03, 08), SRV failures (SPA02, 06, 10), TSV failures (SPA12), and multiple failures (SPA04). Each requires that the generic steam line B system state tree be quantified using conditional probabilities. The generic secondary pressure control system state tree for steam line B is given as Figure 3.2.

3.3.2 Feedwater System State Trees

A second potential contributor to a PTS transient is the overall feedwater supply system. Excessive feedwater (via either the main feedwater supply system, the emergency feedwater supply system, or both) can result in a rapid cooldown of the primary system and lead to a potential PTS condition. The main feedwater system of the Oconee-1 plant consists of two main feedwater pumps, three condensate booster pumps, and three hotwell pumps. Main feedwater (MFW) flow is typically controlled via a feedwater control

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Table 3.1. Secondary pressure control system states

---

1. A. EVENT/STATE - Turbine stop valves on steam line A close.
   B. ABBREV - TSVS A CLOSE
   C. EVENT TYPE: System Response
   D. BRANCH DESCRIPTION:
   STATE 0: Both turbine stop valves on steam line A close.
   STATE 1: At least one TSV on steam line A fails to close.

2. A. EVENT/STATE - Maximum steam line pressure is within designated limits.
   B. ABBREV - MAX SL PRESSURE
   C. EVENT TYPE: Conditioning
   D. BRANCH DESCRIPTION:
   STATE 0: Maximum secondary steam line pressure is $\geq$ 1050 psi. (Demand to both TBVs and SSRVs.)
   STATE 1: Maximum secondary steam line pressure $\geq$ 935 psi but $\leq$ 1050 psi. (Only TBVs demanded.)
   STATE 2: Maximum secondary steam line pressure $\geq$ 935 psi. (No pressure relief demand.)

3. A. EVENT/STATE - Turbine bypass valves on steam line A modulate to maintain steam line pressure.
   B. ABBREV - ALL TBV MODULATE
   C. EVENT TYPE: System Response
   D. BRANCH DESCRIPTIONS:
   STATE 0: Both turbine bypass valves on steam line A modulate to maintain proper steam line pressure.
   STATE 1: At least one TBV fails to function properly resulting in a steam line pressure less than the set point.
   STATE 2: Both TBVs fail to open on demand.

4. A. EVENT/STATE - Secondary side relief valves reseat given pressure reduction below their reseat point.
   B. ABBREV - ALL SSRV RESEAT
   C. EVENT TYPE: System Response
   D. BRANCH DESCRIPTIONS:
   STATE 0: All secondary side relief valves on steam line A reseat given pressure reduction.
   STATE 1: At least one SSRV fails to reseat.

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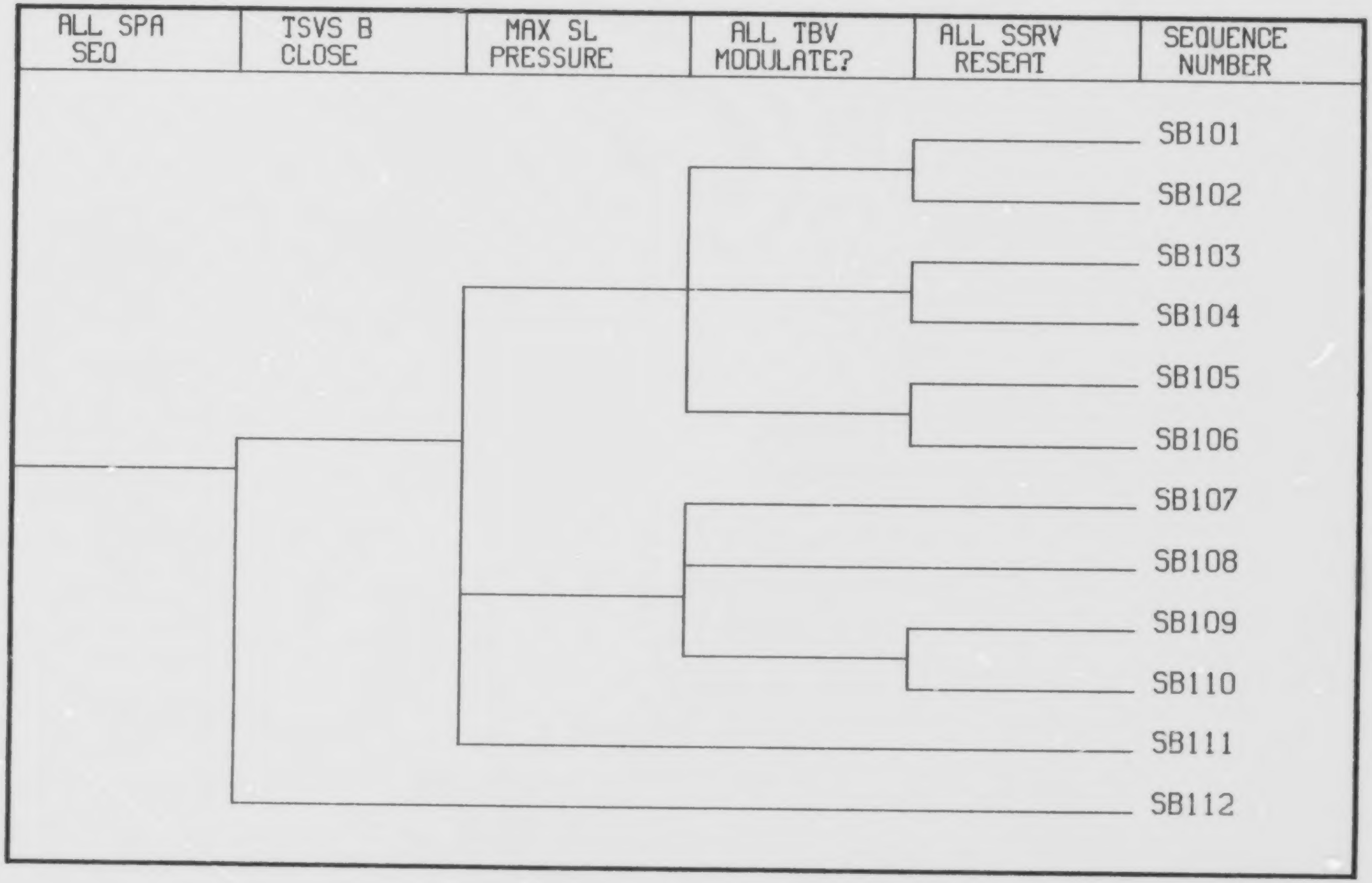
ALL SPA SEQ
TSVS B CLOSE
MAX SL PRESSURE
ALL TBV MODULATE?
ALL SSRV RESEAT
SEQUENCE NUMBER
SB101
SB102
SB103
SB104
SB105
SB106
SB107
SB108
SB109
SB110
SB111
SB112

Figure 3.2 Secondary pressure control (SL B) system state tree; initiator: SPA sequence.

valve, although manual control is possible. The MFW pumps are in turn controlled via the differential pressure across the MFW control valves. The emergency feedwater (EFW) pumps (two motor-driven and one turbine-driven) start on low-MFW pump discharge pressure. Emergency feedwater flow, however, is controlled by the emergency feedwater control valves, which respond to the steam generator (SG) level-sensing system.

Unlike the case for the secondary-side pressure-control state tree, no single conditioning event can be used to simplify the feedwater system state tree. For the feedwater system, three conditioning events are employed, leading to a conditioning event tree. As in the previous section, the conditioning events are included to permit the consideration of both normal and off-normal responses. The relevent conditioning events used consist of the availability of the main feedwater supply system, the transient-induced MFW pump discharge pressure, and the response of the SG level control system to the transient-induced thermal-hydraulic conditions.

The availability of the MFW system is included as a conditioning event to account for those transients whose initiators disable the MFW system and hence preclude any automatic response by the system. The principal example of this type of event is a transient initiated by loss of MFW. The success branch for this conditioning event is defined to be the normal operation of the MFW system immediately following an initiating event. Failure is defined as no MFW flow. This conditioning event also serves to screen transients that begin from initial conditions where the MFW system is not functioning or in operation.

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The second conditioning event employed is a binary branch based on the MFW pump discharge pressure. As noted above, the magnitude of the MFW pump discharge pressure determines the demand for EFW pump actuation. Thus, the second feedwater conditioning event is defined as the minimum MFW pump discharge pressure induced by the transient. State 0 is defined as MFW pump discharge pressure greater than 725 psi (no EFW demand signal), while State 1 is taken to be a pressure less than 725 psi (EFW demanded).

The final conditioning event includes the response of the steam generator level control system. Recognizing the fact that certain transients have the potential for creating false signals in the SG level control system, three relevant branches are defined. The first state (State 0) is one in which the SG level control signals indicate a high-level condition in at least one SG due to transient-induced phenomena. The second state (State 1) is defined as the nominal response of the level control system (i.e., the control signals accurately reflect the condition of the steam generators). The remaining state (State 2) is defined as an indication of a low-level condition in at least one SG due to the transient-induced phenomena.

Combining these three conditioning events results in the feedwater conditioning event tree depicted by Figure 3.3. It should be noted that the high-MFW-pressure case with MFW unavailable has been eliminated because, without the MFW pumps, the discharge pressure is limited by the shutoff head of the condensate booster pumps.

The conditioning event branches give rise to nine feedwater system state trees. However, since the nine trees have many events in common (differing

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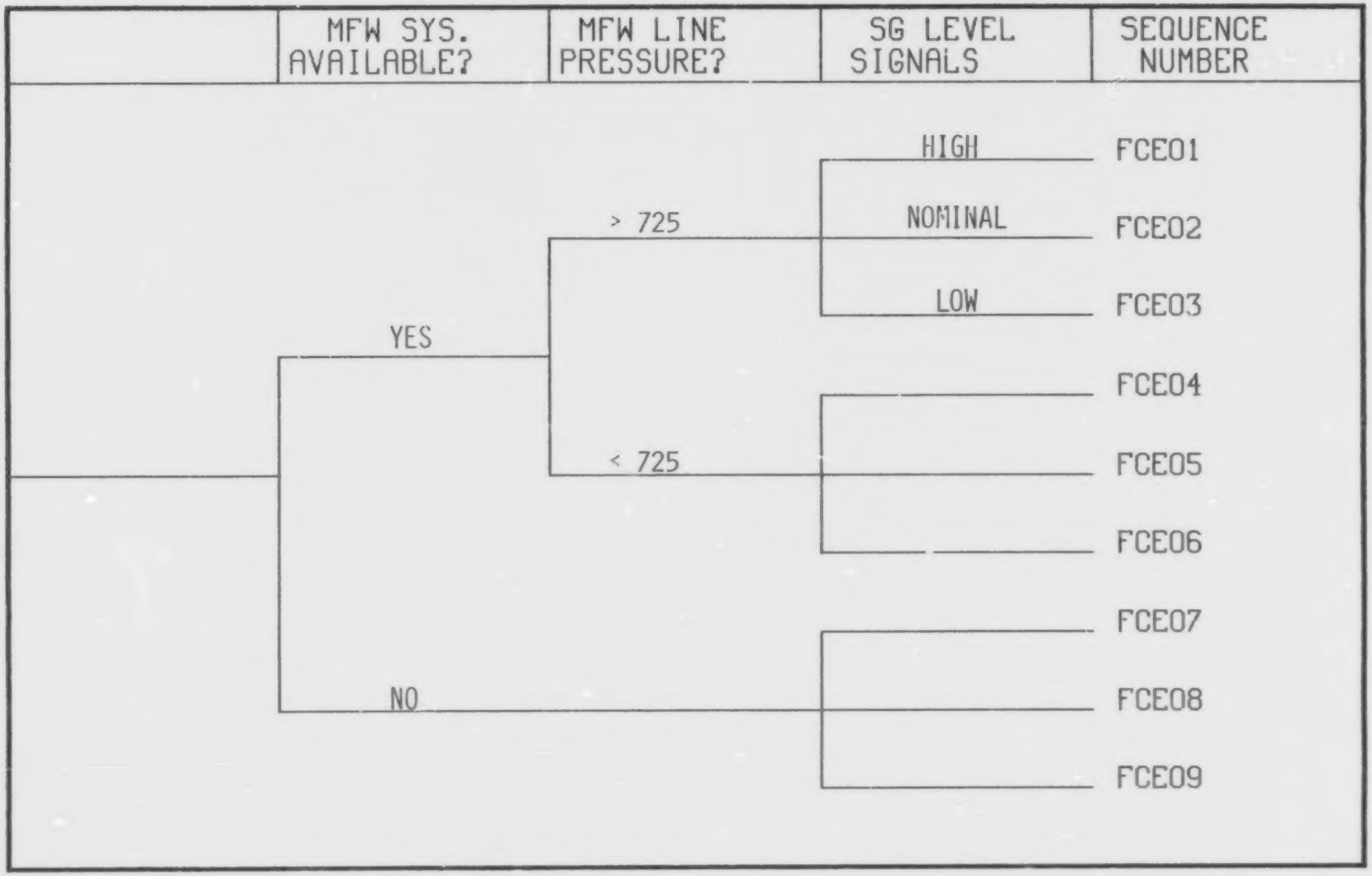
MFW SYS. AVAILABLE?
MFW LINE PRESSURE?
SG LEVEL SIGNALS
SEQUENCE NUMBER
HIGH
NOMINAL
LOW
> 725
< 725
YES
NO
FCE01
FCE02
FCE03
FCE04
FCE05
FCE06
FCE07
FCE08
FCE09

Figure 3.3 Feedwater conditioning event tree; initiator: unspecified.

only in the probability of individual branches), only the significant differences will be called out explicitly in this discussion.

Because the primary concern regarding the feedwater supply is that of excessive feedwater relative to the heat source, the feedwater system trees are constructed on the basis of an implied reduction of the heat source (i.e., a reactor trip). The case of a malfunction resulting in excessive feedwater for a constant heat source will be treated on an exceptional basis later in the discussion. Within this context, the primary event of concern is the success of MFW runback on demand. Four PTS-relevant runback states are of concern: the runback is successful; it is unsuccessful in a single SG resulting in excessive MFW to the SG; it is unsuccessful with regard to both SGs (i.e., a double overfeed); and lastly, the attempted runback results in the trip of the MFW pumps, which for most transients is equivalent to the loss of MFW flow. State 0 is defined as a successful runback wherein the MFW flow to each SG is modulated to maintain the correct SG levels. State 1 is defined as an unsuccessful runback for a single SG with an uncontrolled level increase. State 2 is defined as an unsuccessful runback to each SG resulting in increasing levels, and State 3 is defined as the loss of both MFW pumps as a result of an attempted runback.

The second system of interest concerns the steam generator protection system installed at the Oconee-1 plant. This system is designed to trip the MFW pumps if the level in either SG reaches 90% of the operating range. This level is inferred from Δp sensors in each steam generator. Since the 90% level is beyond the normal operational range, this event is only

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relevant for those branches indicative of overfeeds. The success branch is defined as the trip of both MFW pumps, given that the 90% level has been reached, and the failure branch is defined as the failure to trip at least one MFW pump, given that the 90% level has been reached.

In addition to the response of the MFW system, the response of the emergency feedwater system is also relevant to the PTS analyses. The first response of interest is whether the EFW pumps start, given a demand for EFW. The success branch, denoted by State 0, is defined as the start of at least one (of three) EFW pumps. Two different failure states are distinguished: State 1, failure of a manual start of the EFW pumps; and State 2, failure of all EFW pumps to start due to the unavailability of their emergency feedwater system. This distinction is made because the EFW system can be manually restored, which will be considered in the operator action trees.

The final system response concerns the potential for EFW overfeeds, given the success of EFW pump actuation. Three possible flow-control branches are distinguished. State 0 describes the condition whereby EFW flow to both SGs is modulated to maintain the correct SG water level. For State 1, the EFW flow to a single SG is greater than that required to maintain the appropriate water level, and for State 2, the EFW flow is greater than that required for water level maintenance to both SGs. Table 3.2 summarizes the branch definition for all FW events.

Combining these four system responses under the assumptions of the first feedwater conditioning event end-state (FCE01)--i.e., MFW system operational, MFW pump discharge pressure > 725 psia, and high SG level

Table 3.2. Feedwater system states

---

1. A. EVENT/STATE - MFW system is functioning.
   B. ABBREV - MFWS AVAILABLE?
   C. EVENT TYPE: Conditioning
   D. BRANCH DESCRIPTION:
      STATE 0: At least one MFW pump remains in operation given the initiator.
      STATE 1: Neither MFWP is available. [Condensate system remains available however.]

2. A. EVENT/STATE - Main feedwater line pressure within specified limits.
   B. ABBREV - MFW LINE PRESSURE?
   C. EVENT TYPE: Conditioning
   D. BRANCH DESCRIPTION:
      STATE 0: MFW line pressure $\geq$ 725 psi.
      STATE 1: MFW line pressure $\geq$ 725 psi.

3. A. EVENT/STATE - Apparent GS level signals induced by transients.
   B. ABBREV - SG LEVEL SIGNALS
   C. EVENT TYPE: Conditioning
   D. BRANCH DESCRIPTIONS:
      STATE 0: SG level signals indicate high level (trip level) in at least one SG due to transients-induced phenomena.
      STATE 1: SG level signals accurately reflect the condition of the SGs.
      STATE 2: SG level signals indicate low-level (override) in at least one SG due to transient-induced phenomena.

4. A. EVENT/STATE - Main feedwater system runback state following initiator.
   B. ABBREV - MFWS RUNBACK?
   C. EVENT TYPE: System Response
   D. BRANCH DESCRIPTIONS:
      STATE 0: MFW system runback is successful, and the MFW flow to the SG is modulated to maintain the appropriate levels.
      STATE 1: Runback is unsuccessful and results in an overfeed to 1 flow to the other is modulated to maintain the SG. MFW maintains appropriate level in the unaffected SG.
      STATE 2: Runback is unsuccessful and results in an overfeed to both SGs.
      STATE 3: Runback is unsuccessful and results in the MFW/condensate systems being rendered inoperable temporarily.

5. A. EVENT/STATE - Main feedwater pumps trip on high steam generator level.
   B. ABBREV - MFW PUMPS HLSG TRIP
   C. EVENT TYPE: System Response
   D. BRANCH DESCRIPTIONS:
      STATE 0: System trips both MFWP; given level set point is reached in at least one SG.
      STATE 1: System fails to trip; at least one MFWP given level set point is reached in at least one SG.

6. A. EVENT/STATE - Emergency feedwater pumps start.
   B. ABBREV - EFWPs START
   C. EVENT TYPE: System Response
   D. BRANCH DESCRIPTIONS:
      STATE 0: At least one EFWP (of the two MDEFWP and 1 TDEFWP) starts upon demand.
      STATE 1: No EFW pump starts due to EFW pump controls being in manual.
      STATE 2: No EFW pump starts due to EFW system unavailability.

7. A. EVENT/STATE - Emergency feedwater flow to steam generators controlled or SG level set point.
   B. ABBREV - EFW FLW CNTRL/LEVEL
   C. EVENT TYPE: System Response
   D. BRANCH DESCRIPTIONS:
      STATE 0: EFW flow to both SGs is nominal. The EFW flow to each SG is modulated to maintain the appropriate SG level.
      STATE 1: EFW flow to 1 SG is greater than that required to maintain SG level. Flow to unaffected SG is modulated to maintain the appropriate level.
      STATE 2: EFW flow to both SGs is no greater than that required to maintain SG level.

---

signals--yields the system state tree given in Figure 3.4. For this system state tree, the EFW system is only demanded if the MFW pumps are tripped (either due to high SG level or runback difficulties). Similarly, using the assumption of FCE02, Figure 3.5 can be constructed. The major differences in structure involve the EFW pump activation event of the successful MFW runback branch. Figures 3.6 through 3.10 depict the system state trees resulting from the remaining conditioning event assumptions.

3.3.3 Primary Loop Coolant Flow System State Trees

Whereas both the secondary-side pressure-control systems and the feedwater supply systems influence the fluid conditions seen by the pressure vessel indirectly (via the steam generator), the coolant flow patterns in the primary loops directly impact pressure and temperature. Of concern to the PTS analysis are both the normal primary coolant flow systems (i.e., the status of the reactor coolant pumps and the core vent valves) and the emergency core cooling systems (ECCS). For the Oconee-1 plant, however, the status of the reactor coolant pumps (RCPs) is governed by operator action, so discussion of possible changes of state of the RCPs will be deferred to the section on operator actions. Similarly, because of the intimate connection between the status of the RCPs and the functioning of the vent valves, discussion of the vent valve operation will also be deferred. The remaining systems of concern are the ECC systems [the high-pressure injection system (HPI), the core flood tanks, and the low-pressure injection system (LPI)].

The HPI system is of concern primarily because of the system capability to maintain a high RCS pressure or subsequently to repressurize to a

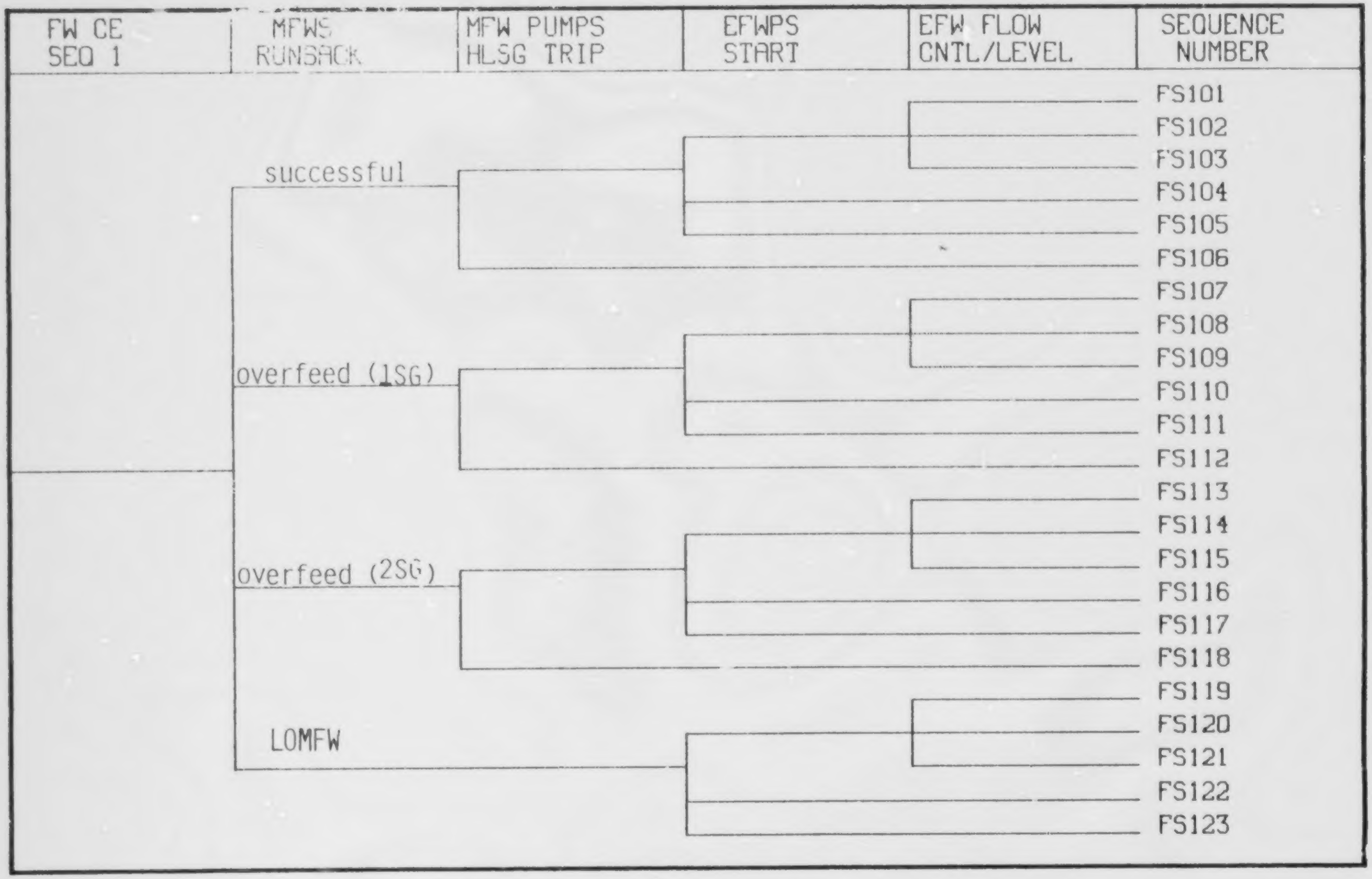
FW CE SEQ 1
MFWS RUNBACK
MFW PUMPS HLSG TRIP
EFWPS START
EFW FLOW CNTL/LEVEL
SEQUENCE NUMBER
successful
overfeed (1SG)
overfeed (2SG)
LOMFW
FS101
FS102
FS103
FS104
FS105
FS106
FS107
FS108
FS109
FS110
FS111
FS112
FS113
FS114
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Figure 3.4 Feedwater system state tree; initiator: FCE sequence 1.

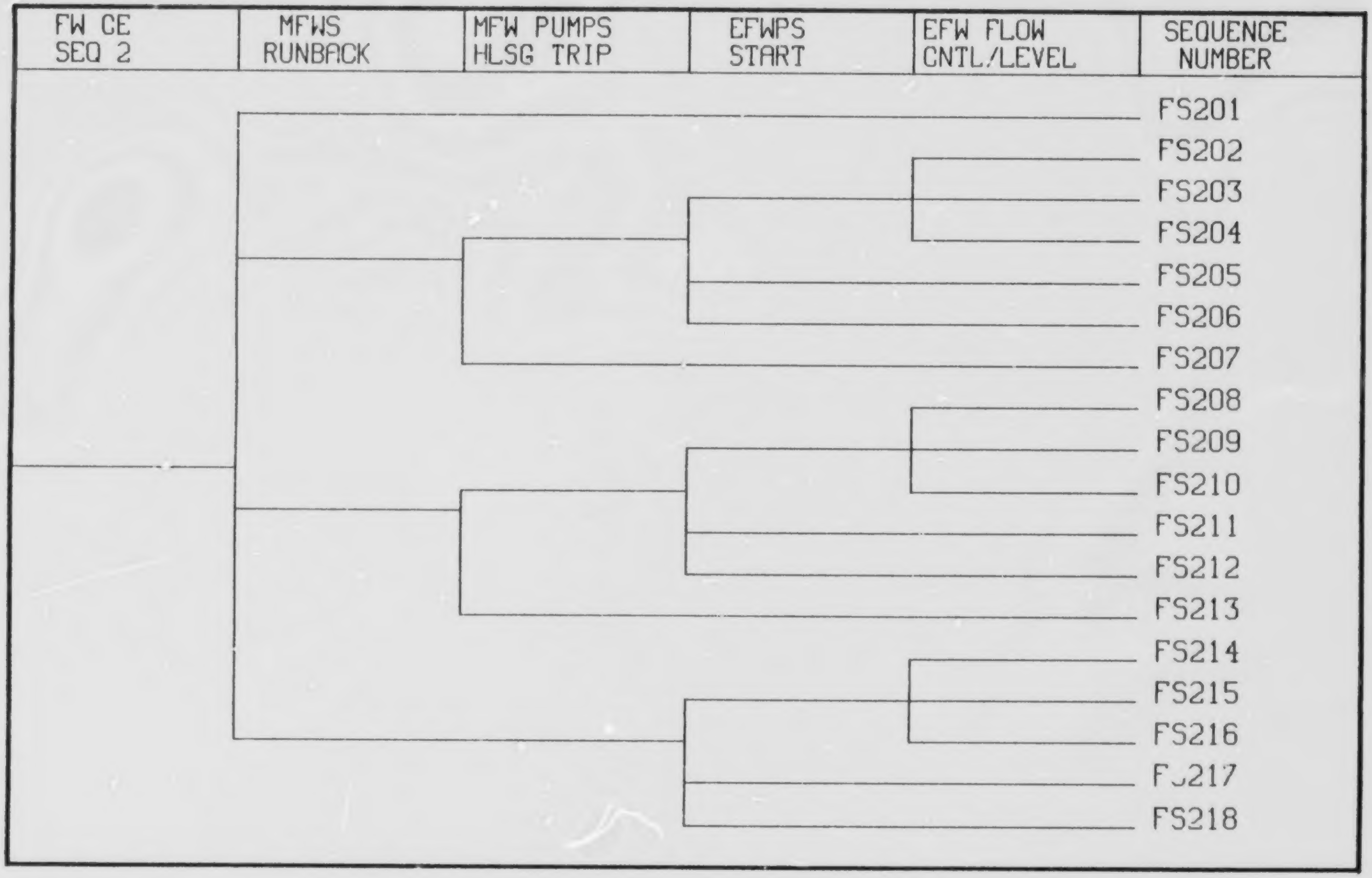

FW CE SEQ 2
MFWS RUNBACK
MFW PUMPS HLSG TRIP
EFWPS START
EFW FLOW CNTL/LEVEL
SEQUENCE NUMBER
FS201
FS202
FS203
FS204
FS205
FS206
FS207
FS208
FS209
FS210
FS211
FS212
FS213
FS214
FS215
FS216
FS217
FS218

Figure 3.5 Feedwater system state tree; initiator: FCE sequence 2.

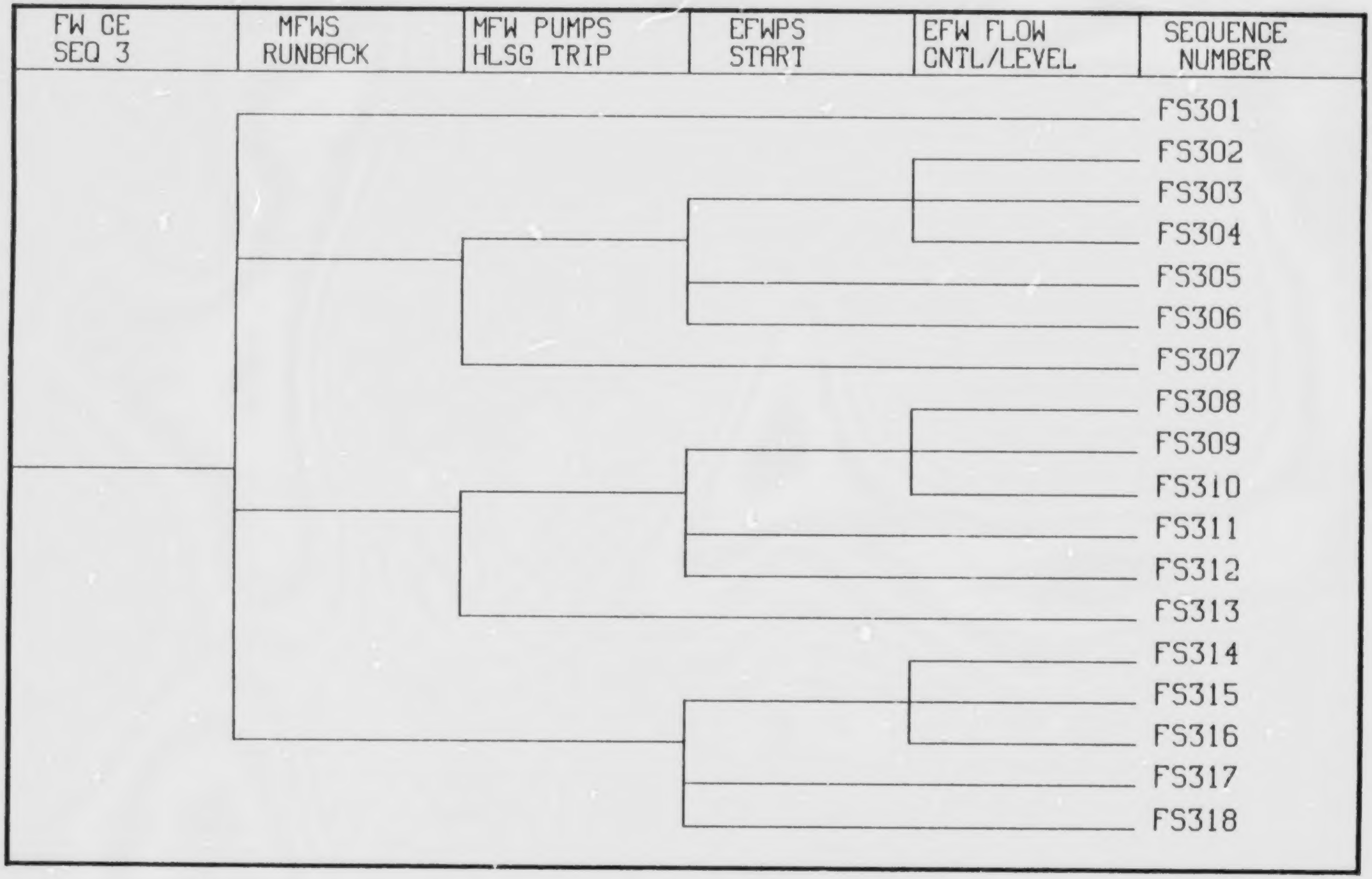
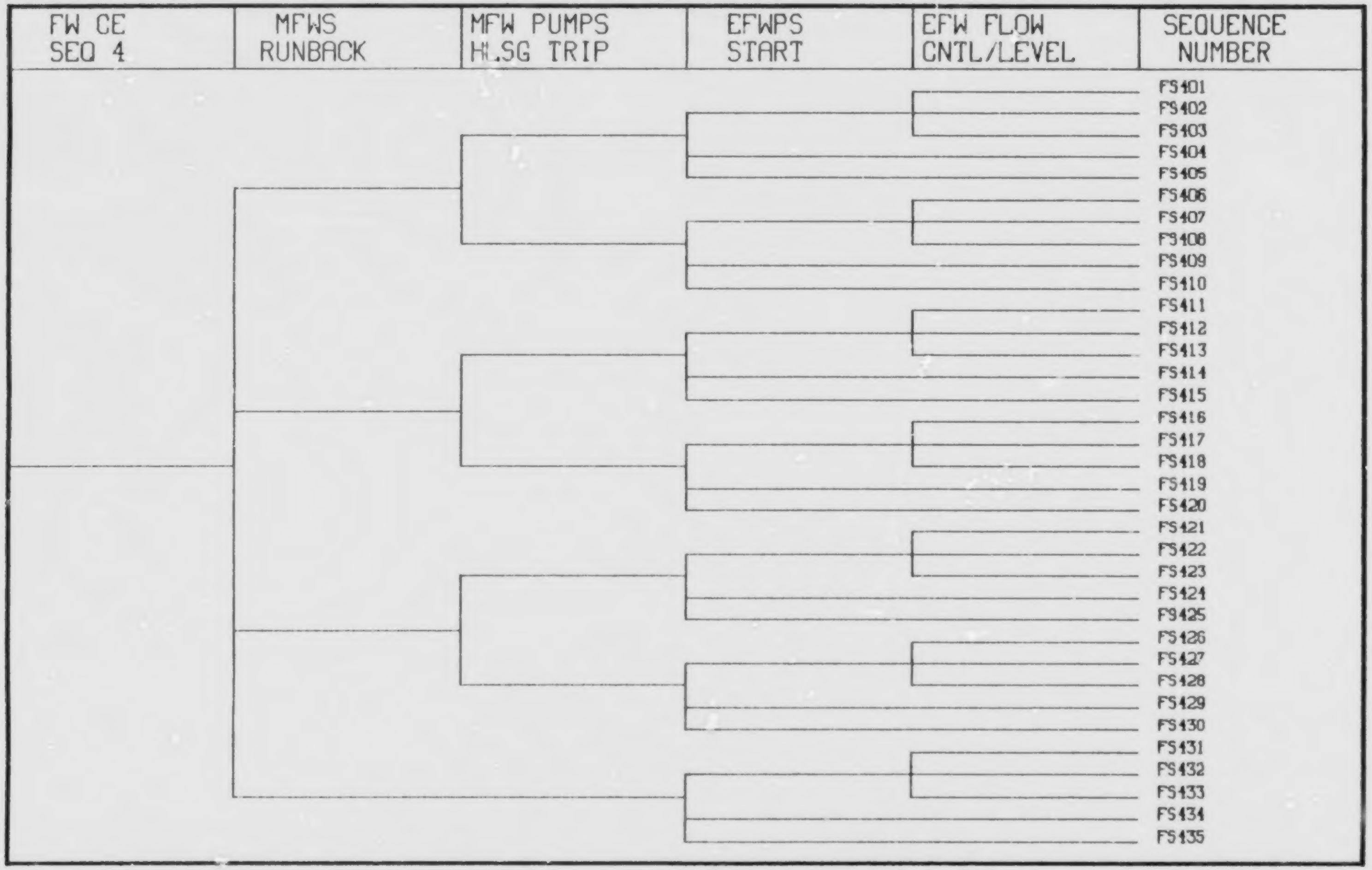

FW CE SEQ 3
MFWS RUNBACK
MFW PUMPS HLSG TRIP
EFWPS START
EFW FLOW CNTL/LEVEL
SEQUENCE NUMBER
FS301
FS302
FS303
FS304
FS305
FS306
FS307
FS308
FS309
FS310
FS311
FS312
FS313
FS314
FS315
FS316
FS317
FS318

Figure 3.6 Feedwater system state tree; initiator: FCE sequence 3.


FW CE
SEQ 4
MFWS
RUNBACK
MFW PUMPS
H.SG TRIP
EFWPS
START
EFW FLOW
CNTL/LEVEL
SEQUENCE
NUMBER
FS401
FS402
FS403
FS404
FS405
FS406
FS407
FS408
FS409
FS410
FS411
FS412
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FS434
FS435

Figure 3.7 Feedwater system state tree; initiator: FCE sequence 4.

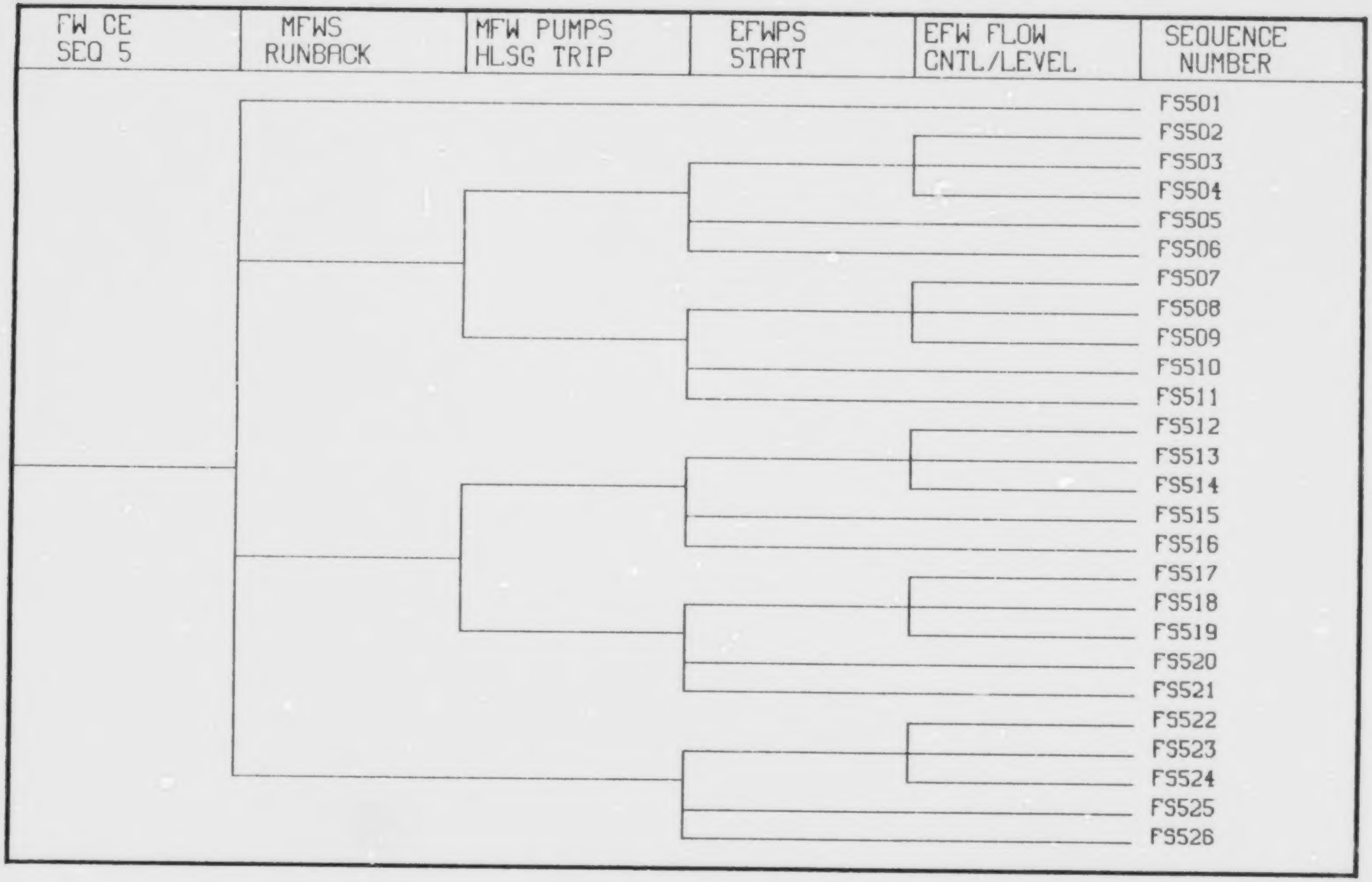
FW CE
SEQ 5
MFWS
RUNBACK
MFW PUMPS
HLSG TRIP
EFWPS
START
EFW FLOW
CNTL/LEVEL
SEQUENCE
NUMBER
FS501
FS502
FS503
FS504
FS505
FS506
FS507
FS508
FS509
FS510
FS511
FS512
FS513
FS514
FS515
FS516
FS517
FS518
FS519
FS520
FS521
FS522
FS523
FS524
FS525
FS526

Figure 3.8 Feedwater system state tree; initiator: FCE sequence 5.

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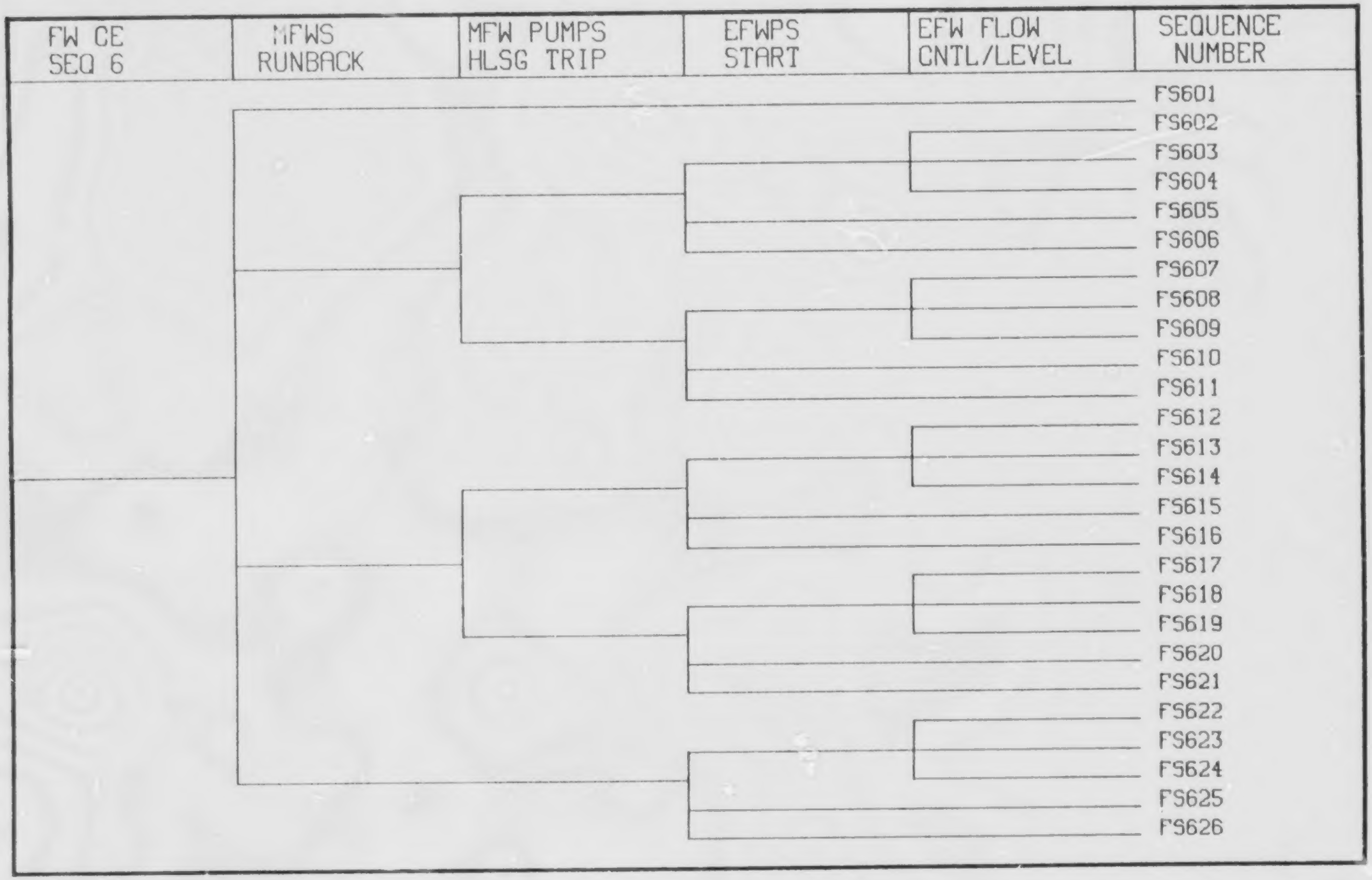
FW CE
SEQ 6
MFWS
RUNBACK
MFW PUMPS
HLSG TRIP
EFWPS
START
EFW FLOW
CNTL/LEVEL
SEQUENCE
NUMBER
FS601
FS602
FS603
FS604
FS605
FS606
FS607
FS608
FS609
FS610
FS611
FS612
FS613
FS614
FS615
FS616
FS617
FS618
FS619
FS620
FS621
FS622
FS623
FS624
FS625
FS626

Figure 3.9 Feedwater system state tree; initiator: FCE sequence 6.

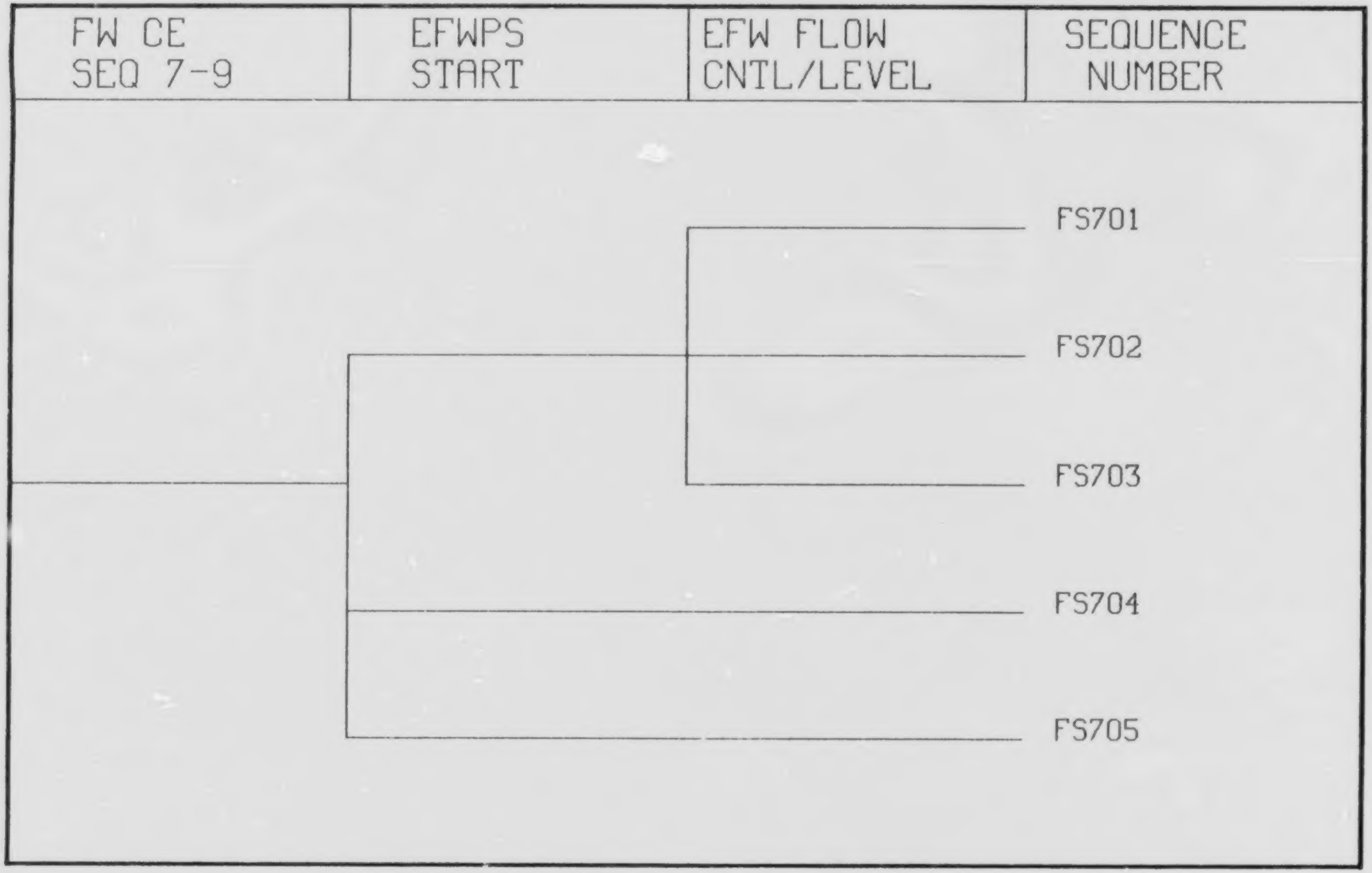
FW CE SEQ 7-9
EFWPS START
EFW FLOW CNTL/LEVEL
SEQUENCE NUMBER
FS701
FS702
FS703
FS704
FS705

Figure 3.10 Feedwater system state tree; initiator: FCE sequences 7, 8, and 9.

high pressure. However, for certain transients (i.e., those with low RC pressure), the temperature effects of the cold HPI fluid may also be an important factor. The core flood tanks and LPI system are of concern for the inverse reason. Although the pressure effect is inherently limited, these systems have the capacity for replacing a large fraction of the primary coolant inventory with "cold" fluid. Moreover, both of these systems inject fluid directly into the downcomer, whereas the HP injection point is 16 ft upstream from the cold leg nozzle.

Since the demand signals for the various ECC systems are based on primary system pressure, the appropriate conditioning parameter is the minimum reactor coolant system pressure reached during the transient. Four possible discriminants are used: State 0 - minimum RC pressure > 1500 psi (no ECCS demand); State 1 - RCS pressure between 610 and 1500 psi (only HPI demanded); State 2 - RCS pressure between 200 and 610 psi (both HPI and CFT demanded); and lastly, State 3 - RCS pressure less than 200 psi (all three systems, HPI, LPI, and CFT, demanded). This conditioning event serves to simplify the construction of the initiator trees by restricting the number of event branches that must be considered.

Given a demand for the HPI system, the next system response of interest is whether the system functions as designed. A binary branch is defined for this event. The success state is defined as transfer of all three HPI pumps to the injection mode and coolant delivery to the RCS. The failure branch is defined as failure of all HPI pumps to deliver coolant, given a demand. As noted in the above discussion, the principal PTS concern in the ECC systems is the success state (i.e., the system functions as designed).

The failure of these systems to operate properly upon demand generally reduces the severity of PTS phenomena. However, the failure branches are included to account for the potential of operator error upon restoration (i.e., overcompensation) because of the generally high reliability of these systems.

The branch point regarding the core flood tanks is defined similarly. The success state is defined as the discharge of at least one core flood tank into the downcomer upon demand. The failure state is the failure of both flood tanks to discharge. The LPI branch is also defined similarly to the previous two branches. The success state is defined as at least one LPI pump's operating and coolant being delivered to the downcomer. The corresponding failure state is that no low-pressure injection occurs.

The system state tree comprising these four events is depicted in Figure 3.11. Table 3.3 summarizes the definitions of the four events.

3.3.4 Primary Pressure Control System State Tree

The PTS concern with the primary-system pressure-control systems (PORV and RVs) is centered on the potential for the creation of loss-of-coolant situations and hence the triggering of ECC systems (if not already running). In particular, the relief valves serve as a mechanism for the removal of heat energy directly from the primary loop.

The appropriate conditioning event for the primary pressure-control system state tree is the extent of the demand for pressure relief as indicated by the maximum RCS pressure attained in the primary loop during the course of the transient. Three states are categorized. State 0 is defined as the

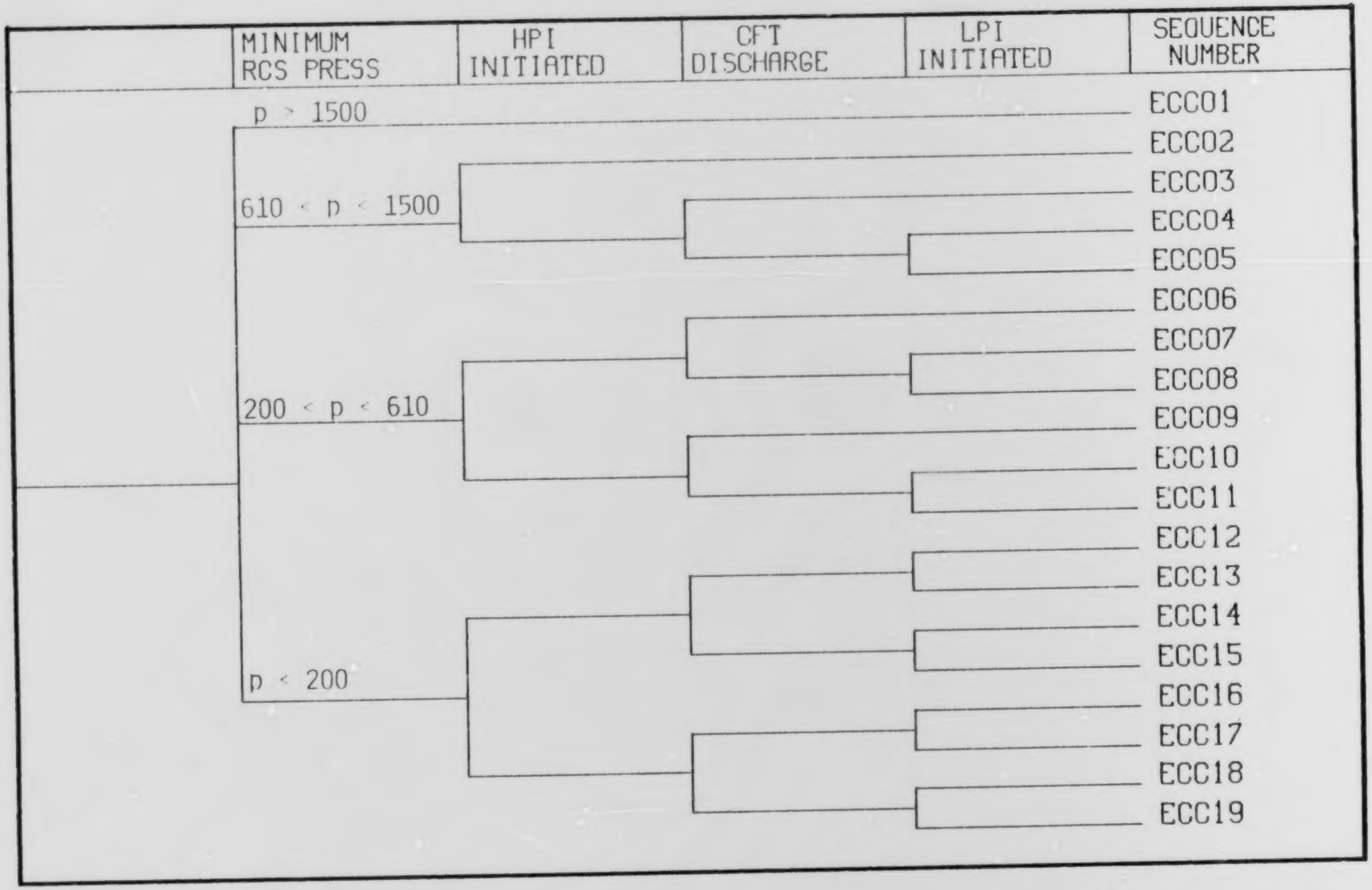
MINIMUM RCS PRESS
HPI INITIATED
CFT DISCHARGE
LPI INITIATED
SEQUENCE NUMBER
p > 1500
610 < p < 1500
200 < p < 610
p < 200
ECC01
ECC02
ECC03
ECC04
ECC05
ECC06
ECC07
ECC08
ECC09
ECC10
ECC11
ECC12
ECC13
ECC14
ECC15
ECC16
ECC17
ECC18
ECC19

Figure 3.11 Primary loop ECCS system state tree; initiator: unspecified.

Table 3.3. Primary ECCS system state tree

---

1. A. EVENT/STATE - Minimum reactor coolant system pressure is within specified limits.
   B. ABBREV - MINIMUM RCS PRESS
   C. EVENT TYPE: Conditioning
   D. BRANCH DESCRIPTION:
      STATE 0: Minimum RCS pressure $\geq$ 1500 psia
      STATE 1: Minimum RCS pressure $\geq$ 610 psia
      STATE 2: Minimum RCS pressure $\leq$ 200 psia
      STATE 3: Minimum RCS pressure $\leq$ 200 psia

2. A. EVENT/STATE - High-pressure injection system activates upon demand.
   B. ABBREV - HPI INITIATED
   C. EVENT TYPE: System Response
   D. BRANCH DESCRIPTION:
      STATE 0: High-pressure injection system injects into reactor coolant system (at least one HPI pump transfers to injection mode).
      STATE 1: No high-pressure injection.

3. A. EVENT/STATE - Core flood tanks discharge inventory to reactor coolant system.
   B. ABBREV - CFT DISCHARGE
   C. EVENT TYPE: System Response
   D. BRANCH DESCRIPTIONS:
      STATE 0: Core flood tanks discharge upon demand.
      STATE 1: Core flood tanks fail to discharge upon demand.

4. A. EVENT/STATE - Low-pressure safety injection into reactor coolant system initiated.
   B. ABBREV - LPI INITIATED
   C. EVENT TYPE: System Response
   D. BRANCH DESCRIPTIONS:
      STATE 0: Low-pressure injection occurs (at least one LP pump operates).
      STATE 1: Low-pressure injection system fails to deliver cooling water.

---

condition in which the maximum RCS pressure remains less than 2450 psi (the nominal PORV set point) and hence there is no demand for pressure relief. State 1 is defined as the condition in wich reactor coolant pressure is between 2450 psi and 2500 psi (the SRV set point), so that only the PORV is demanded. Finally, State 2 is defined as the maximum RCS pressure's being greater than or equal to 2500 psi, so that both PORV and SRVs are demanded.

For those conditioning states wherein the PORV is demanded, the next relevant event is the operability of the PORV. This is, in effect, a two part event. The first is the actual functioning of the PORV represented by a binary event in which the success state is that the PORV lifts upon demand and the failure state is the failure of the PORV to open upon demand. The second part of this event is a conditioning event related to the status of the block valve downstream of the POV. This is included as part of the system state tree, because, if closed, operation of the PORV upon demand will not provide any pressure relief. Again, a binary branch is used, with success defined as the block valve's being open and failure by its being closed.

Given a sufficient pressure relief demand, the next event is the operation of the safety relief valves. The relevent question is whether the relief valves, having opened on demand, reseat when the primary pressure is reduced. If this fails to occur, the plant experiences a transient-induced loss-of-coolant event. The first reseat question concerns the SRVs, and success is defined as the closure, upon pressure reduction, of all the SRVs that have opened. Failure is defined as the failure of at least one SRV

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to reseat. The implicit assumption contained in this analysis is that if the SRVs are demanded, they will open with a probability of 1.

The rationale for discriminating between a stuck-open PORV and a stuck-open SRV is linked to the possible corrective operator actions. The former is an isolable loss-of-coolant accident (LOCA), whereas the later is not.

These five events comprise the primary pressure control system state tree depicted by Figure 3.12. Table 3.4 summarizes the branch definitions specified in the above discussion.

3.4 Quantification of the Functional System State Trees

One of the primary purposes for introducing the concept of a conditioning event was to permit the quantification of each of the four functional system state trees (secondary pressure control, feedwater supply, primary ECC systems, and primary pressure control) separately. Once each is quantified, the initiator-specific event trees are constructed by combining the four functional trees into an overall system state tree and by selecting the appropriate set of conditioning events for the given initiator. This section discusses the numerical values used to quantify each of the four functional system state trees.

Probability values were estimated, when possible, from operational information, which was as close to Oconee-specific as possible. If Oconee-specific information was available, then it was used. If this was not the case, B&W-specific and finally PWR-specific operational information was employed. Additional information was obtained from IEEE-500, the NREP Generic Data Base screening values, as well as from other sources.

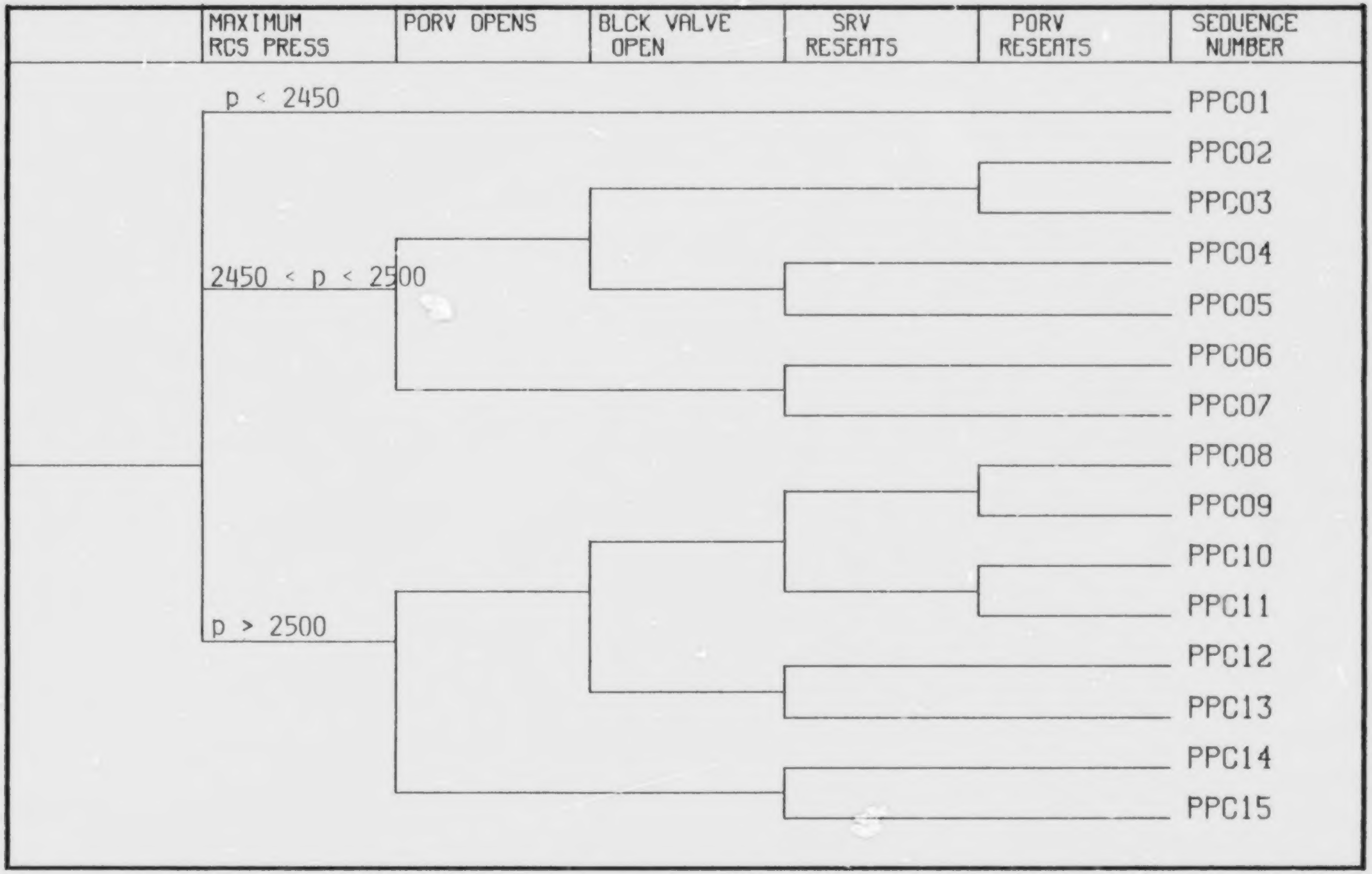
MAXIMUM RCS PRESS
PORV OPENS
BLCK VALVE OPEN
SRV RESEATS
PORV RESEATS
SEQUENCE NUMBER
p < 2450
2450 < p < 2500
p > 2500
PPC01
PPC02
PPC03
PPC04
PPC05
PPC06
PPC07
PPC08
PPC09
PPC10
PPC11
PPC12
PPC13
PPC14
PPC15

Figure 3.12 Primary pressure control system state tree; initiator: unspecified.

Table 3.4. Primary pressure control system state tree

---

1. A. EVENT/STATE - Maximum reactor coolant system pressure is within specified limits.
   B. ABBREV - MAXIMUM RCS PRESS
   C. EVENT TYPE: Conditioning
   D. BRANCH DESCRIPTION:
   STATE 0: Maximum RCS pressure < 2450 psia
   STATE 1: Maximum RCS pressure > 2450 psia
   STATE 2: Maximum RCS pressure > 2500 psia

2. A. EVENT/STATE - Power operated relief valve opens, given RCS pressure > 2450 psi.
   B. ABBREV - PORV OPENS
   C. EVENT TYPE: System Response
   D. BRANCH DESCRIPTION:
   STATE 0: PORV opens at 2450 psi to relief RCS pressure.
   STATE 1: PORV fails to open (fails closed).

3. A. EVENT/STATE - Block valve downstream of PORV is open.
   B. ABBREV - BLCK VALVE OPEN
   C. EVENT TYPE: Conditioning
   D. BRANCH DESCRIPTIONS:
   STATE 0: Block valve is initially open.
   STATE 1: Block valve is initially closed (PROV disabled).

4. A. EVENT/STATE - Safety relief valve reseats, given pressure reduction below reseat pressure.
   B. ABBREV - SRV RESEATS
   C. EVENT TYPE: System Response
   D. BRANCH DESCRIPTIONS:
   STATE 0: Safety relief valve reseats, given pressure reduction.
   STATE 1: Safety relief valve fails to reseat, given pressure reduction.

5. A. EVENT/STATE - Power-generated relief valve reseats, given pressure reduction.
   B. ABBREV - PORV RESATS
   C. EVENT TYPE: System Response
   D. BRANCH DESCRIPTIONS:
   STATE 0: Power operated relief valve reseats, given a sufficient pressure reduction.
   STATE 1: PORV fails to reseat, given pressure reduction.

3.4.1 Quantification of the Secondary Pressure-Control System State Tree

3.4.1.1 Turbine Stop Valves on Main Steam Line (SL) A Close/Demand (TSVs A Close)

STATE 0: P = 1.0 - 9.5E - 0.4 = 0.999 (1-P(State 1))

STATE 1: With regard to the TSVs, all PWR licenses event reports (LERs) were reviewed for turbine stop valve failures. In only one event (NSIC 092449 at Turkey Point 3) was a total failure of all turbine stop valves identified. However, several instances of individual TSV failures to close on turbine trip signal were identified. Based on these data, an estimate for State 1 (at least one TSV fails to close) of 9.5E-04 was used.

3.4.1.2 All TBVs Modulate (Demand for Pressure Relief to Turbine Bypass System Results in the Appropriate Configuration)

STATE 0: All TBVs on SL A modulate to maintain proper steam line pressure.

$$P = 1.0 - \sum_{i=1}^{2} P(\text{STATE}_{i}) = .999$$

STATE 1: At least one TBV fails to modulate from its maximum open position. Oconee Units 1 through 3 have never had an individual failure of these valves to close during a general shutdown (including non specific reactor trip) or startup. Based on 1978-1980 operating experience, there have been approximately 10 shutdowns and startups per plant year. Assuming a bypass valve failure on the next shutdown, one can calculate a bounding failure probability as:

1/[no. of observed demands + 1]
= 1/[4 valves/plant) x (10 shutdowns + 10 startups/plant year) x (Oconee 1, 2, 3, lifetime through 1981) + 1]
= 1/[4 x 20 x 24.2 yrs + 1]
= 5 x 10^{-4}/demand/valve

Thus the probability of failure to open either of the two valves on steam line A is taken as 10^{-3}/Demand.

STATE 2: Both TBVs on steam line A fail to modulate on demand. The analysis for State 1 is applicable to the first valve, and the conditional probability of the second valve failure, given that the first valve failed is taken to be 0.1, is 1.0 x 10^{-4}/Demand.

3.4.1.3 All SSSV Reseat (At Least One Secondary-Side Safety Valve Fails to Reseat, Given All Eight Open and Subsequent Pressure Reduction)

STATE 0: [All SSSVs reseat on pressure reduction]

$$P = 1.0 - P(\text{State}_1) = .92$$

STATE 1: [At least one SSRV fails to reseat upon pressure reduction below the set point.] The probability of a single passive safety valve failing to reseat is assumed to be 1.0 x 10^{-2}. Since each steam line has eight safety valves, the probability is just 8 x 10^{-2}.

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3.4.1.4 Modifications of Secondary Pressure-Control Branches for SL B Due to Conditioning Events/Steam Line A Events

Turbine Stop Valves on SL A Fail To Close

The only event relative to SL B assumed to be conditional on the closure of the TSVs on SL A is the closure of the TSVs on SL B. A conditional probability of failure of 0.1 is assumed in this case to include potential common-course failure coupling.

STATE 0: $P = 0.9$

STATE 1: $P = 0.1$

Turbine Bypass Valves on SL A Fails to Modulate (Fails Open)

A conditional probability of 0.1 for the same failure mode on SL B is assumed. The failure probability for the alternate failure mode is assumed unaffected.

STATE 0: $P = 1-\sum_{i=1}^{2} P(\text{State}_i) \cong 0.9$

STATE 1: $P = 1.0 \times 10^{-1}$

STATE 2: $P = 1.0 \times 10^{-4}$

Turbine Bypass Valves on SL A Fail Closed

A conditional probability of 0.1 for the same failure mode on SL B is assumed. The failure probability for the alternate failure mode is assumed unaffected.

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STATE 0: $P = 1-\sum_{i=1}^{2} P(\text{State}_{i}) \cong 0.9$

STATE 1: $P = 1.0 \times 10^{-3}$

STATE 2: $P = 1.0 \times 10^{-1}$

Secondary Side Safety Valves on SL A Fail to Reseat

A conditional probability (attributable to identical valves, maintenance operations, calibration, etc.) of 0.1 is assumed.

STATE 0: $P = 9.0 \times 10^{-1}$

STATE 1: $P = 1.0 \times 10^{-1}$

Using these assumptions, the overall secondary pressure control system state tree given by Figure 3.13 was constructed.

3.4.2 Quantification of the Feedwater System State Trees

The quantification of the feedwater system state trees will be discussed relative to the following set of conditioning parameters:

a. MFW pumps are running,

b. MFW line pressure remains greater than 725 psia as long as the MFW pumps operate, and

c. The apparent SG level signals are nominally correct.

This set of conditions corresponds to feedwater conditioning event end-state FCE 02 on Figure 3.3.

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3.4.2.1 MFWS Runback (Main Feedwater System Runback Results in a Given Configuration)

STATE 0: Runback of MFW is successful.

$$P = 1.0 - \sum_{i=1}^{3} \text{State}_{i}$$

STATE 1: Runback is unsuccessful; overfeed to either SG. "The Oconee Pressurized Thermal Shock Evaluation" (DPC-RS-1001, January 1982) identifies four failures to runback following a reactor trip:

Unit 2, September 17, 1974

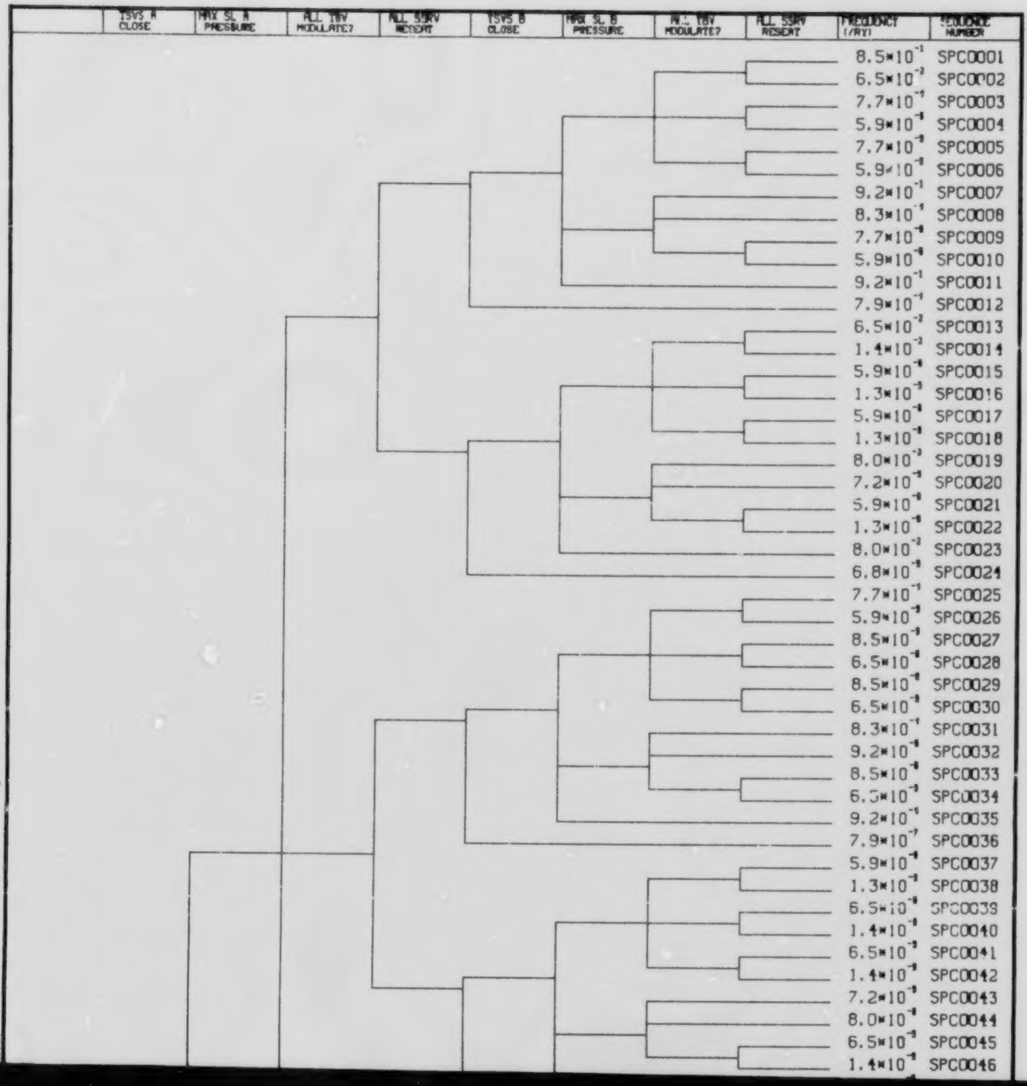
Unit 2, April 30, 1974

Unit 2, January 30, 1980

Unit 3, March 14, 1980

During the time period considered in the Oconee study (24 plant-years), approximately 144 reactor trips occurred. This results in a failure to runback of roughly 4/144 = 2.8 - 10^{-2}/demand.

STATE 2: Runback is unsuccessful, resulting in an overfeed to both SGs. Since a detailed failure analysis was not done for this branch, it is calculated by assuming a conditional probability of 0.1 for the second SG. That is, the probability of an overfeed in the second steam generator given an overfeed in the first = 0.1. Thus, the probability


TSVS A CLOSE
MAX SL A PRESSURE
ALL TBV MODULATE?
ALL SSRV RESEAT
TSVS B CLOSE
MAX SL B PRESSURE
ALL TBV MODULATE?
ALL SSRV RESEAT
FREQUENCY (/RY)
SEQUENCE NUMBER
8.5×10^-1 SPC0001
6.5×10^-2 SPC0002
7.7×10^-4 SPC0003
5.9×10^-5 SPC0004
7.7×10^-8 SPC0005
5.9×10^-8 SPC0006
9.2×10^-1 SPC0007
8.3×10^-4 SPC0008
7.7×10^-8 SPC0009
5.9×10^-8 SPC0010
9.2×10^-1 SPC0011
7.9×10^-4 SPC0012
6.5×10^-3 SPC0013
1.4×10^-3 SPC0014
5.9×10^-8 SPC0015
1.3×10^-5 SPC0016
5.9×10^-8 SPC0017
1.3×10^-8 SPC0018
8.0×10^-3 SPC0019
7.2×10^-9 SPC0020
5.9×10^-8 SPC0021
1.3×10^-8 SPC0022
8.0×10^-3 SPC0023
6.8×10^-8 SPC0024
7.7×10^-1 SPC0025
5.9×10^-9 SPC0026
8.5×10^-9 SPC0027
6.5×10^-8 SPC0028
8.5×10^-8 SPC0029
6.5×10^-9 SPC0030
8.3×10^-1 SPC0031
9.2×10^-8 SPC0032
8.5×10^-8 SPC0033
6.5×10^-9 SPC0034
9.2×10^-6 SPC0035
7.9×10^-7 SPC0036
5.9×10^-8 SPC0037
1.3×10^-9 SPC0038
6.5×10^-8 SPC0039
1.4×10^-8 SPC0040
6.5×10^-9 SPC0041
1.4×10^-9 SPC0042
7.2×10^-9 SPC0043
8.0×10^-8 SPC0044
6.5×10^-9 SPC0045
1.4×10^-9 SPC0046

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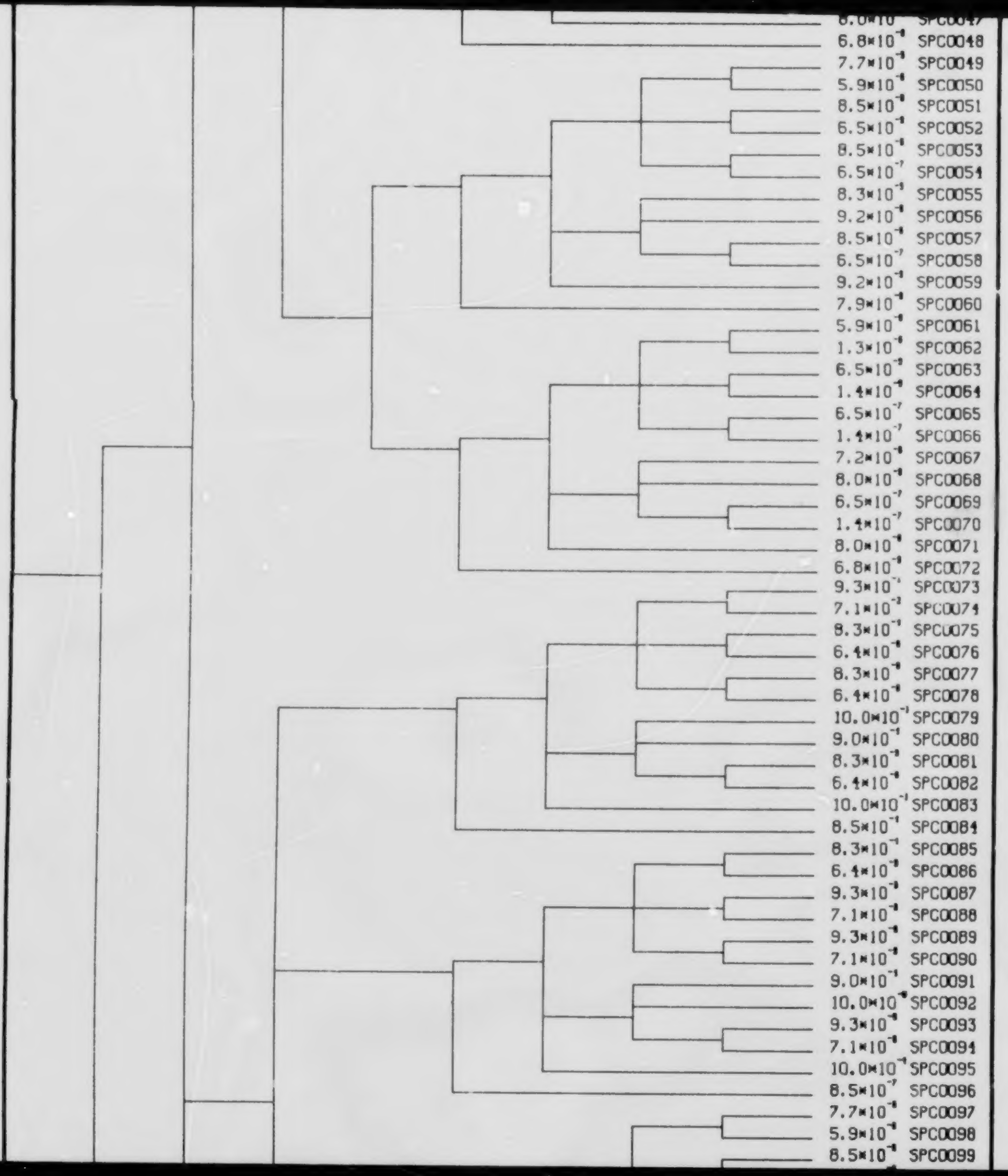
8.0×10 SPC0047
6.8×10^{-8} SPC0048
7.7×10^{-8} SPC0049
5.9×10^{-8} SPC0050
8.5×10^{-8} SPC0051
6.5×10^{-8} SPC0052
8.5×10^{-8} SPC0053
6.5×10^{-7} SPC0054
8.3×10^{-8} SPC0055
9.2×10^{-8} SPC0056
8.5×10^{-8} SPC0057
6.5×10^{-7} SPC0058
9.2×10^{-8} SPC0059
7.9×10^{-8} SPC0060
5.9×10^{-8} SPC0061
1.3×10^{-8} SPC0062
6.5×10^{-8} SPC0063
1.4×10^{-8} SPC0064
6.5×10^{-7} SPC0065
1.4×10^{-7} SPC0066
7.2×10^{-8} SPC0067
8.0×10^{-8} SPC0068
6.5×10^{-7} SPC0069
1.4×10^{-7} SPC0070
8.0×10^{-8} SPC0071
6.8×10^{-8} SPC0072
9.3×10^{-1} SPC0073
7.1×10^{-3} SPC0074
8.3×10^{-1} SPC0075
6.4×10^{-8} SPC0076
8.3×10^{-8} SPC0077
6.4×10^{-8} SPC0078
10.0×10^{-1} SPC0079
9.0×10^{-1} SPC0080
8.3×10^{-8} SPC0081
6.4×10^{-8} SPC0082
10.0×10^{-1} SPC0083
8.5×10^{-1} SPC0084
8.3×10^{-1} SPC0085
6.4×10^{-8} SPC0086
9.3×10^{-8} SPC0087
7.1×10^{-8} SPC0088
9.3×10^{-8} SPC0089
7.1×10^{-8} SPC0090
9.0×10^{-1} SPC0091
10.0×10^{-8} SPC0092
9.3×10^{-8} SPC0093
7.1×10^{-8} SPC0094
10.0×10^{-1} SPC0095
8.5×10^{-7} SPC0096
7.7×10^{-8} SPC0097
5.9×10^{-8} SPC0098
8.5×10^{-8} SPC0099

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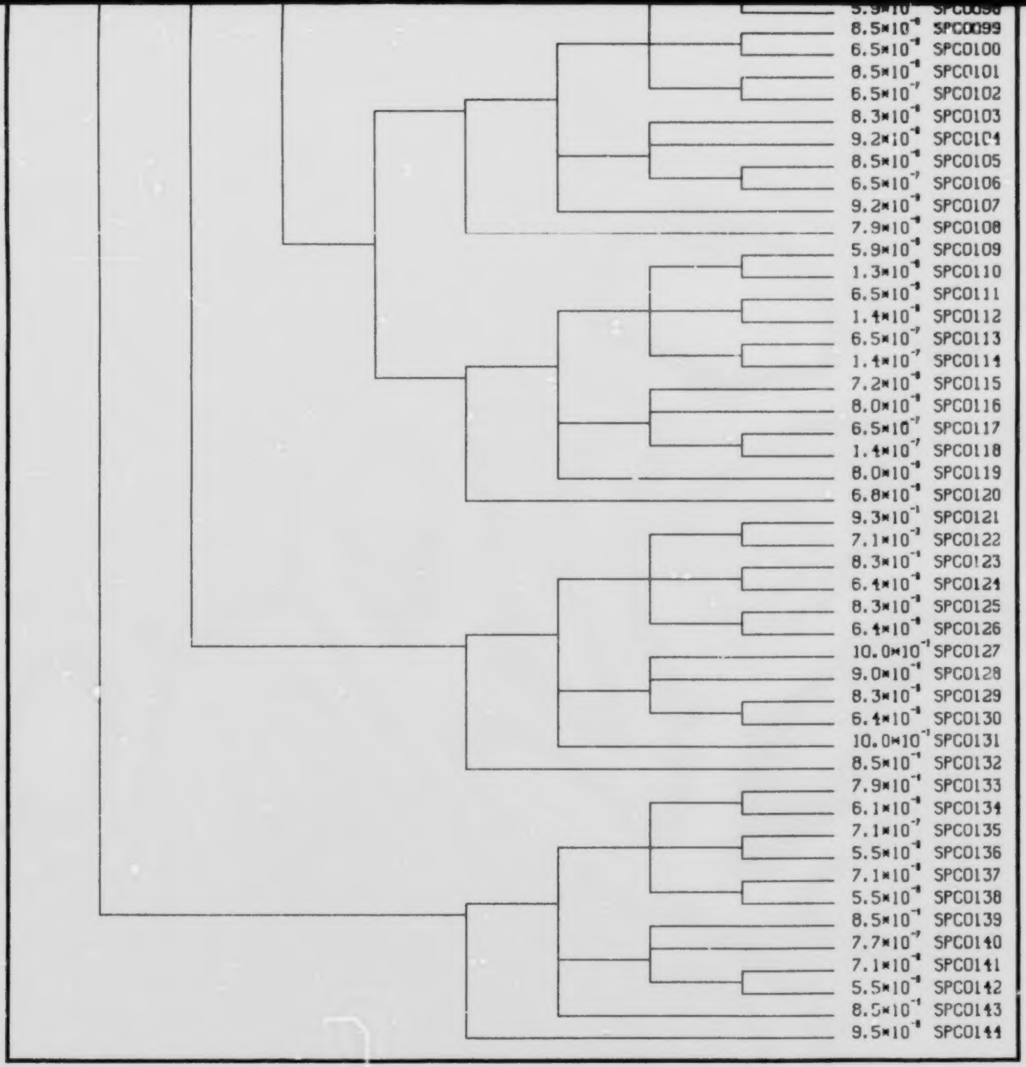
5.9×10 SPC0098
8.5×10^{-8} SPC0099
6.5×10^{-8} SPC0100
8.5×10^{-8} SPC0101
6.5×10^{-7} SPC0102
8.3×10^{-8} SPC0103
9.2×10^{-8} SPC0104
8.5×10^{-8} SPC0105
6.5×10^{-7} SPC0106
9.2×10^{-8} SPC0107
7.9×10^{-8} SPC0108
5.9×10^{-8} SPC0109
1.3×10^{-8} SPC0110
6.5×10^{-3} SPC0111
1.4×10^{-8} SPC0112
6.5×10^{-7} SPC0113
1.4×10^{-7} SPC0114
7.2×10^{-8} SPC0115
8.0×10^{-8} SPC0116
6.5×10^{-7} SPC0117
1.4×10^{-7} SPC0118
8.0×10^{-8} SPC0119
6.8×10^{-8} SPC0120
9.3×10^{-1} SPC0121
7.1×10^{-3} SPC0122
8.3×10^{-1} SPC0123
6.4×10^{-8} SPC0124
8.3×10^{-8} SPC0125
6.4×10^{-8} SPC0126
10.0×10^{-1} SPC0127
9.0×10^{-1} SPC0128
8.3×10^{-8} SPC0129
6.4×10^{-8} SPC0130
10.0×10^{-1} SPC0131
8.5×10^{-1} SPC0132
7.9×10^{-1} SPC0133
6.1×10^{-8} SPC0134
7.1×10^{-7} SPC0135
5.5×10^{-8} SPC0136
7.1×10^{-8} SPC0137
5.5×10^{-8} SPC0138
8.5×10^{-1} SPC0139
7.7×10^{-7} SPC0140
7.1×10^{-8} SPC0141
5.5×10^{-8} SPC0142
8.5×10^{-1} SPC0143
9.5×10^{-8} SPC0144

Figure 3.13 Secondary pressure control system state tree.

of this branch is given by $1.4 \times 10^{-2} \times 0.1 \times 2$ combination $= 2.8 \times 10^{-3}$.

STATE 3: Runback is unsuccessful, resulting in a trip of the MFW pumps. NUREG-0560 identifies five events at B&W plants (through April 4, 1979, the date of IE Bulletin 79-05A) in which feedwater was lost subsequent to a reactor trip. If the Oconee-1 trip rate is assumed to be a typical value for B&W plants (6 trips/year x 30.18 years of operation = 181 trips), this results in a probability of $5/181 = 2.8^{-2}$. This estimate does not include instrument bus failures that result in a simultaneous reactor trip and a loss of MFW. $P = 2.8 \times 10^{-2}$.

3.4.2.2 MFW Pumps HLGS Trip (Failure of the High Steam Generator Signal to Trip Both MFW Pumps Given on Overfeed)

This trip circuit is tested yearly. Assuming a general instrumentation failure rate of 1.0E - 06/h (IEEE-500) results in a failure probability of $1.0E - 06 \times 8760 \times 0.5 = 4.4 \times 10^{-3}$ for the SG experiencing the overfeed. This estimate is appropriate for those branches involving overfeeds to a single SG. For the case of a high-level trip failure, given a double overfeed, an alternate estimate must be made. Each SG has a separate trip system; however, all signals (from either generator) use a common relay. Thus, for branches involving a double overfeed, a value of 1.0×10^{-4}/demand (NPRDS value for relays) is assumed.

STATE 0: [High level protection system trips both MFW pumps]

$P = 1 - P(\text{State}_1) = .996$

STATE 1: [High level protection system fails to trip at least 1 MFW pump]

$P_1 = 4 \times 10^{-3}$

3.4.2.3 EFW Pumps Start (Failure of All EFW Pumps to Start on Demand)

STATE 0: [At least one EFW pump starts on demand.]

$$P = 1 - \sum_{i=1}^{2} P(\text{State}_i) = .993$$

STATE 1: [Failure of EFW pumps to start on demand - system in manual control.] The EFW system is assumed to be in manual control (i.e., for testing maintenance, etc.) approximately 4 h/mo. This results in a probability to start automatically on demand of $4/(30 \times 24) = 5.6 \times 10^{-3}$ $P = 5.6 \times 10^{-3}$.

STATE 2: [Failure of EFW pumps to start on demand - system is unavailable.] NUREG/CR-2497 identified emergency feedwater failures on demand. This value was 1.1×10^{-3}, but it included credit for short-term rectification. Excluding the possibility of rectification (to be included in the operator action trees), the observed value was 1.4×10^{-3}/demand. Oconee observed EFW unavailabilities at the pump level (in 1980 and 1981) are

Oconee	Unavailability	Date
1	turbine pump	5/80

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2	turbine pump	4/81
3	turbine pump	11/80
1	turbine pump	6/81
3	motor-driven pump	5/81
2	motor-driven pump	3/81
2	turbine pump	5/81

For the 17-mo period between May 1980 and September 1981, there were seven unavailabilities, which results in an unavailability per pump of 7/(1 test [demand]/mo x 17 mo x 3 pumps/plant x 3 plants) = 0.05/demand. With only one pump required for success, one can derive a system failure probability of 0.05 (failure of first pump) x 0.3 (failure of third, given failure of first and second) = 1.5 x 10^{-3}/demand. This is consistent with the NUREG/CR-2497 value of 1.4 x 10^{-3}/demand.

3.4.2.4 EFW Flow CNTL/Level (Failure to Modulate EFW Flow to the SGs to Maintain SG Water Level)

STATE 0: [EFW flow is modulated to maintain appropriate levels in both SGs.)

$$P = 1.0 - \sum_{i=1}^{2} P(\text{State}_i)$$

STATE 1: [EFW flow is excessive to 1 SG.] The EFW level circuit consists of a level transmitter on each steam generator, an associated E/P converter, and flow control valve. Additional

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instrumentation is provided to change the SG level based on whether or not the RCPs are operating, An alternate level control channel is provided in the event power is lost to the operating channel. It is assumed that accessible instrumentation is tested monthly and that the level changes during operation can be cross-checked among the SG operating and startup range level indications. Based on this, one can roughly estimate the probability of overfill on initiation as the probability of tranmitter failure or converter failure or valve failure (10^{-6}/h x 720 h (level controllers, IEEE-500) + 1 x 10^{-3}/demand (valve failure, WASH-1400) = 1.7 x 10^{-3}/demand for a specified generator. This gives an estimate for either steam generator of 3.4E-03.

3.4.3 Probability Modifications Required for Off-Normal Situations

3.4.3.1 Apparent SG Level Signals Indicate High Level

MFWS Runback: No modification

For a normal runback (e.g., following reactor trip close to full power), the SGs initially contain more inventory than is required. Thus, it is assumed that the integrated control system (ICS) perceives this situation as normal.

MFW Pumps HLSG Trip: No Modification

The probability of the SG protection system functioning is assumed independent of the reading. The net effect of these two assumptions

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however is that trip of the MFW pumps becomes the most likely branch given the conditioning event.

EFW Pumps Start: No Modification

EFW Flow CNTL/Level: No Modifications

Although the same phenomena which produce the apparent high SG level signals provided to the ICS will likely affect the EFW flow control system, the phenomena are assumed to be of short duration. Since this system does not involve trips, normal operation will be recovered when the phenomenon ceases.

3.4.3.2 Apparent SG Level Signals Indicate Low Level

MFWS Runback

The assumption inherent in this conditioning event is that the SG phenomena will result in a low-level that will which will override the runback to the affected steam generator and will, in effect, overfeed the generator.

STATE 0: Given a low SG level control signal in the affected SG, it is assumed that the probability of a normal runback is 0.0.

STATE 1: $P = 1.0 - \sum_{i=1}^{3} P(\text{State}_i) = 0.96$

STATE 2: The probability of an overfeed to a second generator is assumed to be the probability of a single SG overfeed on attempted runback, which is equal to 1.4×10^{-2}.

STATE 3: The probability of a loss of the MFWP on the attempted runback is assumed to be unchanged by this conditioning event and thus equals 2.8×10^{-2}.

MFW Pumps HLSG Trip

Since the SG high-level (HL) protection signals are based on the same type of sensors as the runback signals ($\Delta\rho$ sensors), it is likely that, at least for the duration of the apparent low-level signals, the HL trip system will be inoperative. Thus, for the branch involving a single overfeed, the failure probability of the HL trip system is assumed to be 1. However, for the case of a double overfeed, the unaffected SGs protection system will function. For this case, a failure probability of 4.4×10^{-3} (the same value used for a single protection circuit) is assumed.

EFW Pumps Start (No Modifications)

EFW Flow CNTL/Level

Again, the same phenomena which affect the MFW control system will affect the EFW flow control system. Thus, the EFW to the affected SG will likely be excessive for the duration of the phenomenon. It is assumed that this is of short duration, and since no trips are involved, continued EFW overfeed would require a system failure. Hence, the branch probabilities assigned in the reference case are used.

Using these assumptions, the overall feedwater supply system state tree given by Figure 3.14 was constructed.

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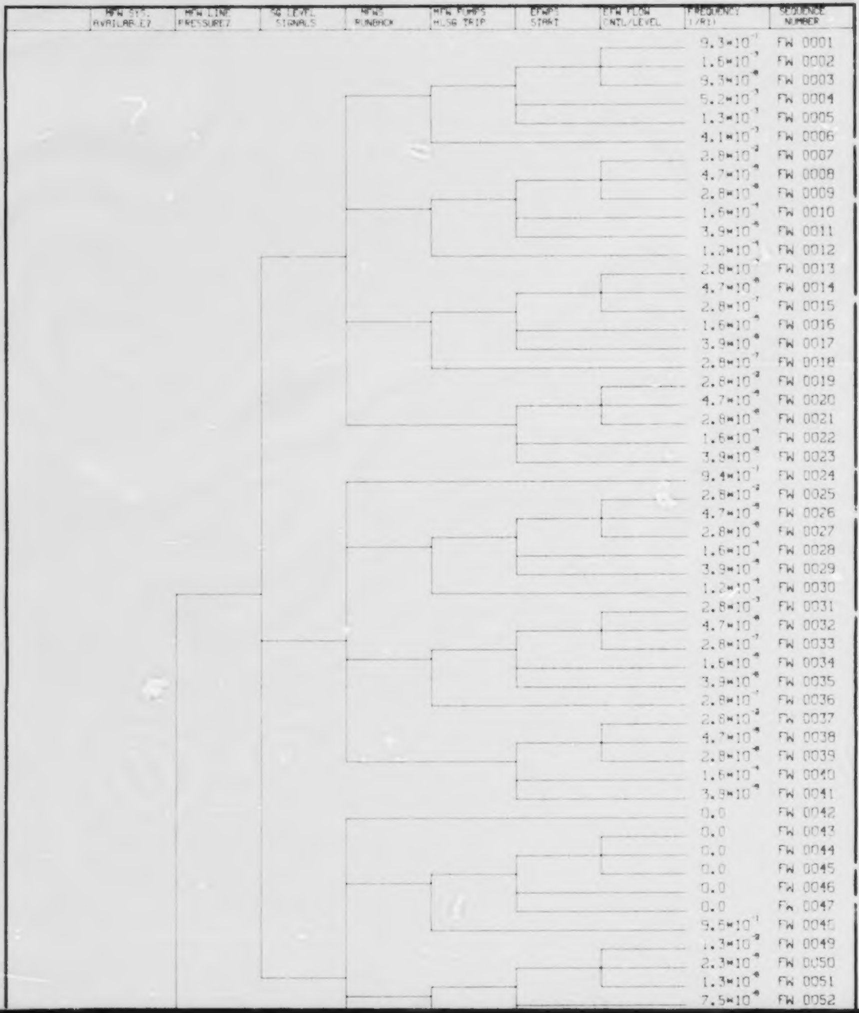
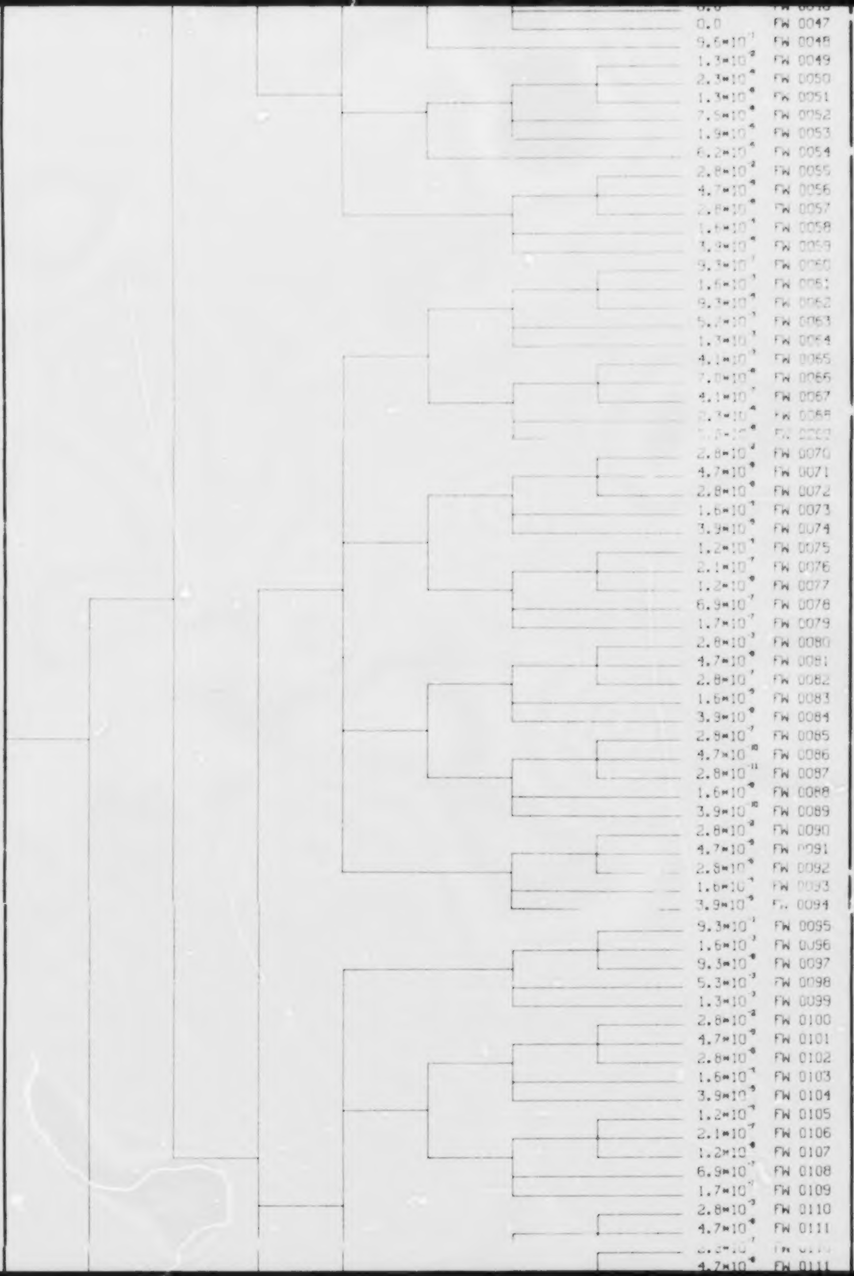

MFW SYS. AVAILABLE?
MFW LINE PRESSURE?
SG LEVEL SIGNALS
MFWS RUNBACK
MFW PUMPS HLSG TRIP
EFWPS START
EFW FLOW CNTL/LEVEL

FREQUENCY (/RY)	SEQUENCE NUMBER
9.3×10^{-1}	FW 0001
1.6×10^{-3}	FW 0002
9.3×10^{-6}	FW 0003
5.2×10^{-3}	FW 0004
1.3×10^{-3}	FW 0005
4.1×10^{-3}	FW 0006
2.8×10^{-2}	FW 0007
4.7×10^{-5}	FW 0008
2.8×10^{-6}	FW 0009
1.6×10^{-4}	FW 0010
3.9×10^{-5}	FW 0011
1.2×10^{-4}	FW 0012
2.8×10^{-3}	FW 0013
4.7×10^{-6}	FW 0014
2.8×10^{-7}	FW 0015
1.6×10^{-5}	FW 0016
3.9×10^{-6}	FW 0017
2.8×10^{-7}	FW 0018
2.8×10^{-2}	FW 0019
4.7×10^{-5}	FW 0020
2.8×10^{-6}	FW 0021
1.6×10^{-4}	FW 0022
3.9×10^{-5}	FW 0023
9.4×10^{-1}	FW 0024
2.8×10^{-2}	FW 0025
4.7×10^{-5}	FW 0026
2.8×10^{-6}	FW 0027
1.6×10^{-4}	FW 0028
3.9×10^{-5}	FW 0029
1.2×10^{-4}	FW 0030
2.8×10^{-3}	FW 0031
4.7×10^{-6}	FW 0032
2.8×10^{-7}	FW 0033
1.6×10^{-5}	FW 0034
3.9×10^{-6}	FW 0035
2.8×10^{-7}	FW 0036
2.8×10^{-2}	FW 0037
4.7×10^{-5}	FW 0038
2.8×10^{-6}	FW 0039
1.6×10^{-4}	FW 0040
3.9×10^{-5}	FW 0041
0.0	FW 0042
0.0	FW 0043
0.0	FW 0044
0.0	FW 0045
0.0	FW 0046
0.0	FW 0047
9.5×10^{-1}	FW 0048
1.3×10^{-2}	FW 0049
2.3×10^{-5}	FW 0050
1.3×10^{-6}	FW 0051
7.5×10^{-6}	FW 0052


0.0 FW 0046
0.0 FW 0047
9.6×10^{-7} FW 0048
1.3×10^{-2} FW 0049
2.3×10^{-4} FW 0050
1.3×10^{-8} FW 0051
7.5×10^{-8} FW 0052
1.9×10^{-4} FW 0053
6.2×10^{-4} FW 0054
2.8×10^{-2} FW 0055
4.7×10^{-8} FW 0056
2.8×10^{-8} FW 0057
1.6×10^{-5} FW 0058
3.9×10^{-8} FW 0059
9.3×10^{-7} FW 0060
1.6×10^{-3} FW 0061
9.3×10^{-4} FW 0062
5.2×10^{-5} FW 0063
1.3×10^{-3} FW 0064
4.1×10^{-3} FW 0065
7.0×10^{-8} FW 0066
4.1×10^{-7} FW 0067
2.3×10^{-4} FW 0068
[illegible] FW 0069
2.8×10^{-2} FW 0070
4.7×10^{-8} FW 0071
2.8×10^{-8} FW 0072
1.6×10^{-4} FW 0073
3.9×10^{-5} FW 0074
1.2×10^{-4} FW 0075
2.1×10^{-7} FW 0076
1.2×10^{-8} FW 0077
6.9×10^{-7} FW 0078
1.7×10^{-7} FW 0079
2.8×10^{-3} FW 0080
4.7×10^{-8} FW 0081
2.8×10^{-7} FW 0082
1.6×10^{-5} FW 0083
3.9×10^{-8} FW 0084
2.9×10^{-7} FW 0085
4.7×10^{-10} FW 0086
2.8×10^{-11} FW 0087
1.6×10^{-8} FW 0088
3.9×10^{-10} FW 0089
2.8×10^{-2} FW 0090
4.7×10^{-9} FW 0091
2.8×10^{-8} FW 0092
1.6×10^{-4} FW 0093
3.9×10^{-5} FW 0094
9.3×10^{-1} FW 0095
1.6×10^{-3} FW 0096
9.3×10^{-8} FW 0097
5.3×10^{-3} FW 0098
1.3×10^{-3} FW 0099
2.8×10^{-2} FW 0100
4.7×10^{-9} FW 0101
2.8×10^{-8} FW 0102
1.6×10^{-4} FW 0103
3.9×10^{-5} FW 0104
1.2×10^{-4} FW 0105
2.1×10^{-7} FW 0106
1.2×10^{-8} FW 0107
6.9×10^{-7} FW 0108
1.7×10^{-7} FW 0109
2.8×10^{-3} FW 0110
4.7×10^{-8} FW 0111
[illegible]
4.7×10^{-8} FW 0111

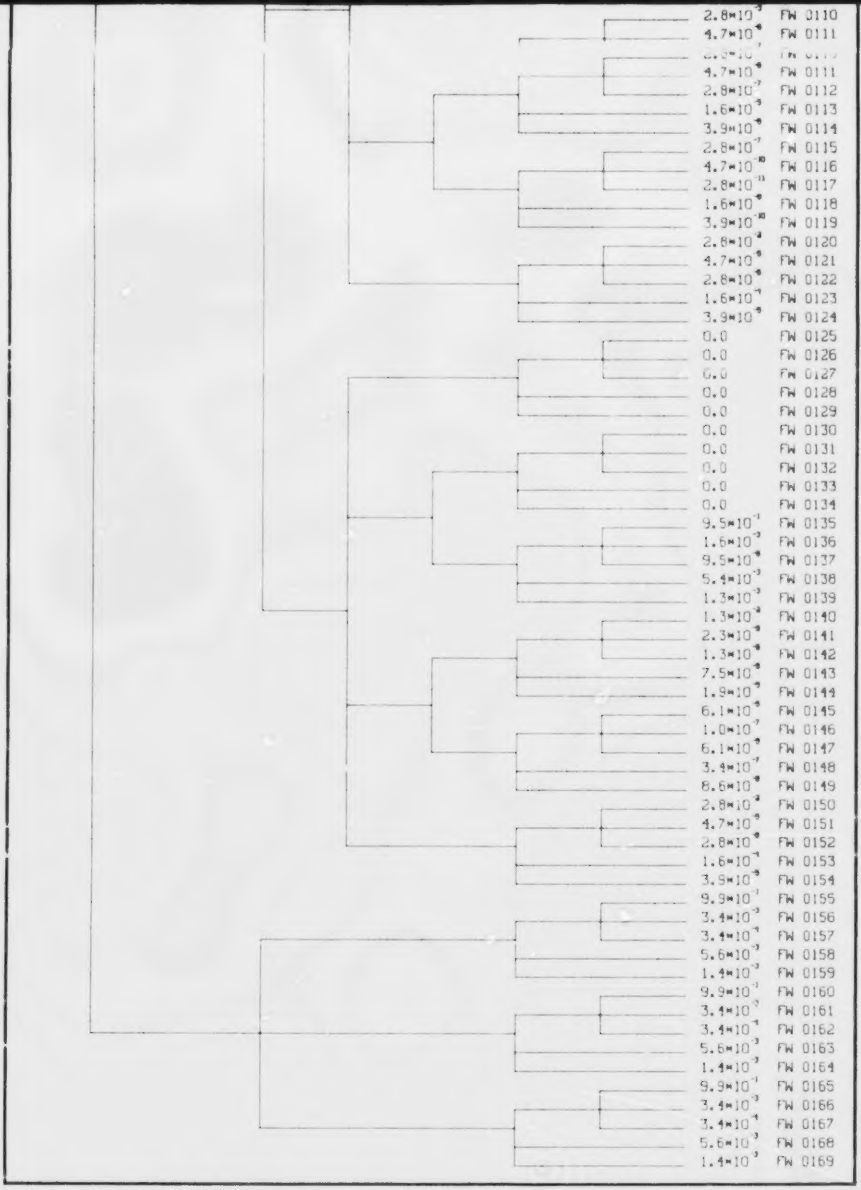

2.8×10^{-5} FW 0110
4.7×10^{-8} FW 0111
[illegible]
4.7×10^{-8} FW 0111
2.8×10^{-7} FW 0112
1.6×10^{-5} FW 0113
3.9×10^{-8} FW 0114
2.8×10^{-7} FW 0115
4.7×10^{-10} FW 0116
2.8×10^{-11} FW 0117
1.6×10^{-8} FW 0118
3.9×10^{-10} FW 0119
2.8×10^{-4} FW 0120
4.7×10^{-5} FW 0121
2.8×10^{-6} FW 0122
1.6×10^{-4} FW 0123
3.9×10^{-5} FW 0124
0.0 FW 0125
0.0 FW 0126
0.0 FW 0127
0.0 FW 0128
0.0 FW 0129
0.0 FW 0130
0.0 FW 0131
0.0 FW 0132
0.0 FW 0133
0.0 FW 0134
9.5×10^{-1} FW 0135
1.6×10^{-3} FW 0136
9.5×10^{-8} FW 0137
5.4×10^{-3} FW 0138
1.3×10^{-3} FW 0139
1.3×10^{-2} FW 0140
2.3×10^{-8} FW 0141
1.3×10^{-8} FW 0142
7.5×10^{-8} FW 0143
1.9×10^{-5} FW 0144
6.1×10^{-5} FW 0145
1.0×10^{-7} FW 0146
6.1×10^{-8} FW 0147
3.4×10^{-7} FW 0148
8.6×10^{-8} FW 0149
2.8×10^{-2} FW 0150
4.7×10^{-5} FW 0151
2.8×10^{-6} FW 0152
1.6×10^{-4} FW 0153
3.9×10^{-5} FW 0154
9.9×10^{-1} FW 0155
3.4×10^{-3} FW 0156
3.4×10^{-4} FW 0157
5.6×10^{-3} FW 0158
1.4×10^{-3} FW 0159
9.9×10^{-1} FW 0160
3.4×10^{-3} FW 0161
3.4×10^{-4} FW 0162
5.6×10^{-3} FW 0163
1.4×10^{-3} FW 0164
9.9×10^{-1} FW 0165
3.4×10^{-3} FW 0166
3.4×10^{-4} FW 0167
5.6×10^{-3} FW 0168
1.4×10^{-3} FW 0169

Figure 3.14 Feedwater system state tree.

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3.4.4 Quantification of the Primary Loop ECCS System State Tree

3.4.4.1 High-Pressure Injection System Activates on Demand

STATE 0: $P = 1.0 - P(\text{State}_1)$

STATE 1: NUREG/CR-2497 provides a probability for HPI system failures on demand of 1.3×10^{-3}/demand. HPI unavailabilities at the pump/train level observed at Oconee between April 1978 and September 1981 (41 plant-months) include:

Oconee	Unavailability	Date
3	train B	11/78
3	train A	4/78
2	pump A	2/81
2	pump B	1/81
2	pump B	11/80
2	pump B	7/80
2	pump B	7/80

Assuming these seven failures to be unconnected (this is, in actuality, probably not so with the four Oconee 2 pump B failures) on both a train and pump basis results in the following unavailability estimates:

$7/(41 \text{ mo} \times 2 \text{ trains/plant} \times 3 \text{ plants}) = 0.028$

and

$7/(41 \text{ mo} \times 3 \text{ pumps/plant} \times 3 \text{ plants}) = 0.019$

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With only one train or pump required for success, one can derive system failure probabilities of 0.028 (first train) x 0.1 (second train x failure given failure of the first train) = 2.8 x 10^{-3}/demand

and

0.019 (first pump) x 0.1 (second pump x failure given failure of the first) x 0.3 (third pump x failure given failure of the first and second) = 6 x 10^{-4}/demand. These values are consistent with the NUREG/CR-2497 value of 1.3 x 10^{-3}/demand, which was used in this analysis.

3.4.4.2 Core Flood Tanks Discharge upon Demand

STATE 0: $P = 1.0 - P(\text{State}_1)$

STATE 1: Since the CFT system is a passive one, the failure probability is assumed to be dominated by the failure of either check valve connecting the CFTs to the downcomer to open on demand. This is assumed to be 4.0 x 10^{-5} for at least one check valve in each line.

3.4.4.3 Low-Pressure Safety Injection into Downcomer Initiated

STATE 0: $P = 1.0 - P(\text{State}_1)$

STATE 1: A value of 4.5 x 10^{-4} for the unavailability of the LPI system (given CFT successful operation) is assumed.

3.4.4.4 Modifications of Primary Loop ECCS Probabilities Due to Conditioning Events and/or Previous Failures

The only system assumed to be affected by either conditioning events or previous failures is the LPI system whose operation is assumed conditional on the success or failure of the core flood tanks. This interdependence arises because both systems flow through a common check valve. Thus, for those branches involving failure of the CFT to inject given a demand, the corresponding LPI probabilities are assumed to be:

STATE 0: 0.75

STATE 1: 0.25

Using these assumptions, the overall primary loop ECCS system state tree given by Figure 3.15 was constructed.

3.4.5 Quantification of the Primary Loop Pressure Control System State Tree

3.4.5.1 PORV Opens on Demand

STATE 0: $1 - P(\text{State}_1) \simeq 1.0$

STATE 1: A value of 1.0×10^{-5} (WASH-1400) was assumed.

3.4.5.2 SRV Reseats upon Demand

STATE 0: $1 - P(\text{State}_1)$

STATE 1: The probability of a single SRV failing to reseat is assumed

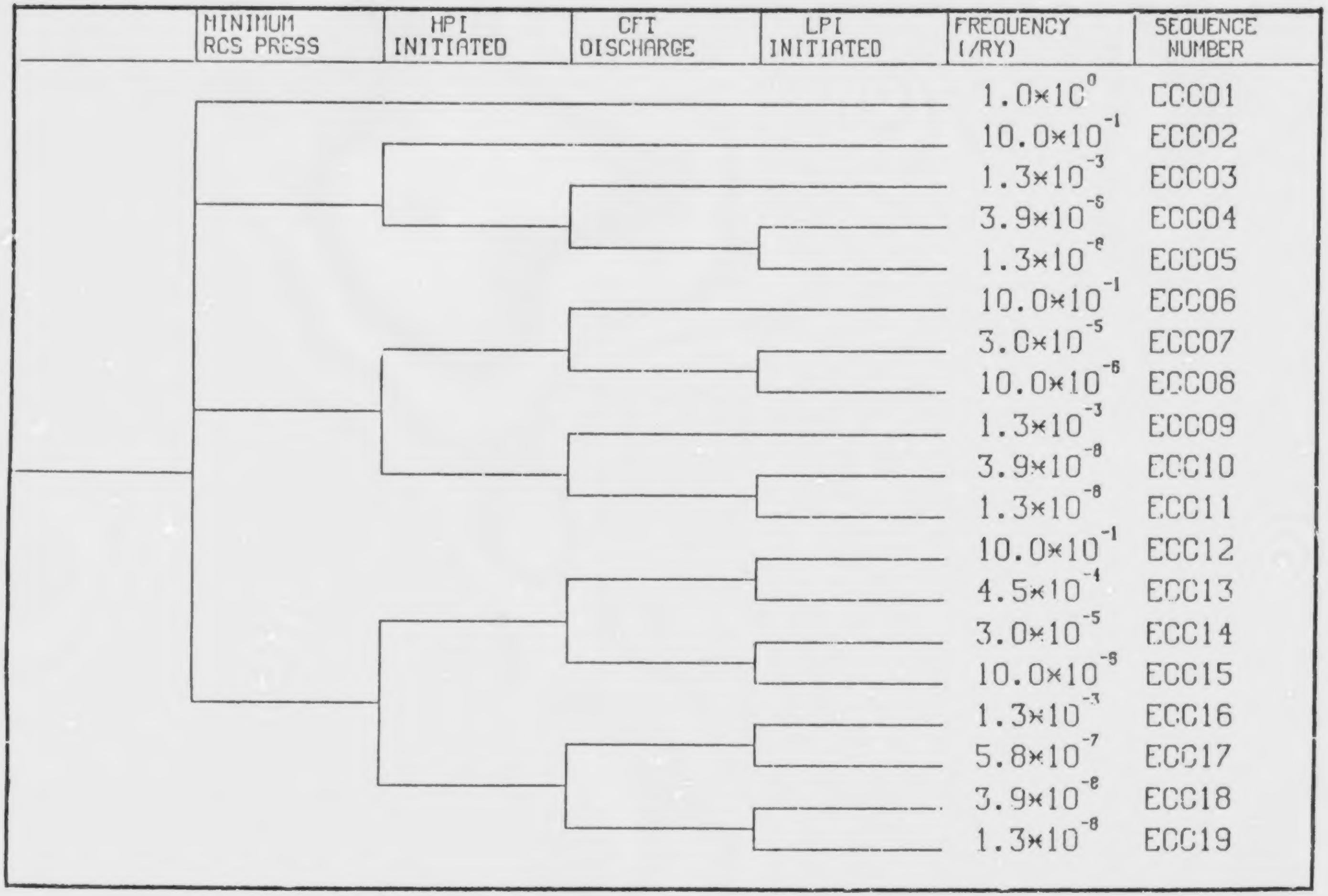
MINIMUM RCS PRESS
HPI INITIATED
CFT DISCHARGE
LPI INITIATED
FREQUENCY (/RY)
SEQUENCE NUMBER
1.0×10^0 ECC01
10.0×10^-1 ECC02
1.3×10^-3 ECC03
3.9×10^-8 ECC04
1.3×10^-8 ECC05
10.0×10^-1 ECC06
3.0×10^-5 ECC07
10.0×10^-6 ECC08
1.3×10^-3 ECC09
3.9×10^-8 ECC10
1.3×10^-8 ECC11
10.0×10^-1 ECC12
4.5×10^-4 ECC13
3.0×10^-5 ECC14
10.0×10^-6 ECC15
1.3×10^-3 ECC16
5.8×10^-7 ECC17
3.9×10^-8 ECC18
1.3×10^-8 ECC19

Figure 3.15 Primary loop ECCS system state tree; initiator: unspecified.

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to be 1.0 x 10^{-2}/demand. Since the Oconee 1 plant has two SRVs, a probability of 2.9 x 10^{-2} is used.

3.4.5.3 PORV Reseats upon Demand

STATE 0: 1 - P(State $_1$) = .973

STATE 1: NUREG-0560 reports four failures of the PORV to close when expected, once it had opened, in approximately 150 demands. This results in a failure probability of 4/15 = 2.7 x 10^{-2}.

Using these assumptions, the overall primary pressure control system state tree given by Figure 3.16 was constructed.

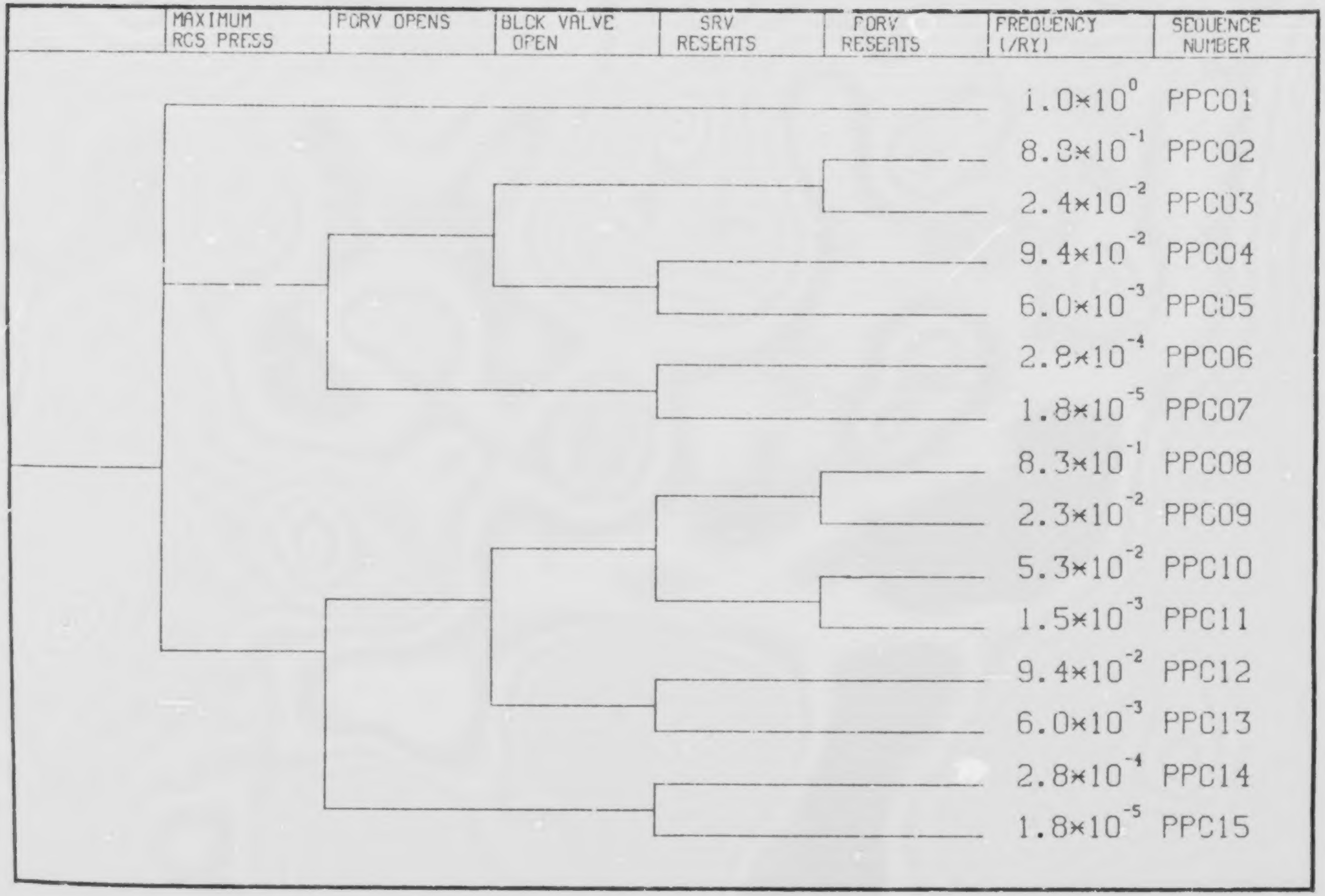
MAXIMUM RCS PRESS
PORV OPENS
BLCK VALVE OPEN
SRV RESEATS
PORV RESEATS
FREQUENCY (/RY)
SEQUENCE NUMBER
1.0×10^{0} PPC01
8.8×10^{-1} PPC02
2.4×10^{-2} PPC03
9.4×10^{-2} PPC04
6.0×10^{-3} PPC05
2.8×10^{-4} PPC06
1.8×10^{-5} PPC07
8.3×10^{-1} PPC08
2.3×10^{-2} PPC09
5.3×10^{-2} PPC10
1.5×10^{-3} PPC11
9.4×10^{-2} PPC12
6.0×10^{-3} PPC13
2.8×10^{-4} PPC14
1.8×10^{-5} PPC15

Figure 3.16 Primary pressure control system state tree; initiator: unspecified.

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3.5 TRANSIENT CATEGORIZATION

To reduce the potentially large number of end-states (6,9335,760) that result from the combination of the four system state trees (SPC, FWS, ECC, and PPC), it is useful to group transients with similar thermal-hydraulic behavior into categories. The individual end state frequencies can be summed to give a category frequency and a representative transient selected to estimate the conditional probability of vessel failure, given the occurrence of the transient. The categories were chosen largely on the basis of the operator action involved, but all were not monotonically increasing in PTS severity. The following categories have therefore been defined:

3.5.1 Secondary Pressure-Control System (SPC)

CATEGORY 0: No significant thermal-hydraulic effect. This category accounts for those transients in which either no malfunctions take place or in which a malfunction does occur but does not affect the course of the transient. An example of the latter case is that of a TBV failure to open on demand, but where the SSRVs operate to limit the secondary side pressure.

CATEGORY 1: This category is appropriate for those transients involving a nonisolable break in a single steam line. Typically, breaks in the steam line upstream of the TSVs, or stuck-open SSRVs, fall in this category.

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CATEGORY 2: This category involves those transients in which an isolable break is present in a single steam line. This category is typified by a stuck open TBV (isolable by the turbine block valves) or a steam line break downstream of the TSVs.

CATEGORY 3: This category involves those transients in which both steam lines are subject to depressurization due to a secondary pressure-control malfunction and neither steam line can be isolated from the break(s). An example of this category is a transient involving a stuck open SSRV on each steam line (perhaps because of a common-cause effect).

CATEGORY 4: This category accounts for those transients in which both steam lines are involved but in which one can be isolated from the break(s). It thus combines malfunctions of Categories 1 and 2.

CATEGORY 5: This category accounts for those transients in which both steam lines are involved and in which both steam lines can be isolated from the break(s). An example of this category is a transient involving a stuck-open TBV on each steam line.

CATEGORY 6: This category is a catchall to account for those transients (end-states) that cannot be characterized

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by Categories 0 through 5 and that must be examined individually to determine the thermal-hydraulic response.

3.5.2 Feedwater System (FWS)

CATEGORY 0: No significant thermal-hydraulic effect. An example of this category would be a transient in which the MFW system is lost on an attempted runback together with the operation of the EFW system as designed.

CATEGORY 1: This category accounts for those transients that involve an unsustained MFW overfeed. That is, an MFW overfeed is terminated by the high SG level trip system.

CATEGORY 2: This category accounts for those transients that involve a sustained MFW overfeed, typically because of a failure of the high-level trip system.

CATEGORY 3: This category accounts for those transients that involve an overfeed by the EFW system. Thus, it also includes those transients in which a MFW overfeed has taken place (either sustained or unsustained) together with an EFW overfeed.

CATEGORY 4: This category accounts for those transients in which all feedwater (MFW and EFW) is lost to the steam generators. An example transient in this category is one in which the MFW system is lost on an attempted runback and the EFW system fails to initiate.

3.5.3 Emergency Core Cooling System (ECCS)

CATEGORY 0: This category accounts for the transients in which no ECCS failures occur. An example of this category is one in which no demand for the ECCS function is made (i.e., primary system pressure remains above the HPI set point of 1500 psi).

CATEGORY 1: This category is comprised of those transients in which the HPI system fails to initiate upon demand. Although HPI failure effectively prohibits system repressurization, the HPI failure can cause a demand for the CFT and LPI systems (which inject directly into the downcomer), which is a potential PTS concern. Moreover, the possibility of operator action to restore the HPI function must be addressed.

CATEGORY 2: This category accounts for multiple failures within the ECC system, such as HPI and CFT failures.

3.5.4 Primary Pressure Control System (PPC)

CATEGORY 0: This category is characterized by no failures within the primary pressure control system (i.e., the PORV and SRVs).

CATEGORY 1: This category involves those breaks in the primary coolant system which can be isolated. An example of this

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is a stuck-open PORV that can be isolated by the block valve downstream.

CATEGORY 2: This category involves those primary system break that are not isolable, such as a stuck-open SRV or a small break in one of the coolant legs.

As a notational convenience, transients are identified by four category numbers, ordered as follows: SPC, FWS, ECC, and PPC. Thus, a transient designated as (1,3,0,1) indicates SPC Category 1, FWS Category 3, ECC Category 0, and PPC Category 1, and can be described as a nonisolable break in one steam line coupled with an EFW overfeed and a failure of the PORV.

3.6 Initiator-Specific System State Tree Construction

To employ the methodology outlined in the previous sections, a initiator-specific system state tree must be constructed for each initiator of interest. This construction consists of specifying conditioning event states appropriate for the initiator for each of the four system state trees, combining the four system state trees into an overall system state tree that describes the asymptotic end-states that remain possible, pruning those end states whose frequency is below the cutoff value (taken as 1.0e-06/RY in this study) from the overall tree, and finally evaluating the effect of operator actions by which the frequency (and in some cases, the possibility) of various end states can be altered. The initiators considered in this study are listed in Table 3.5. For each initiator listed, an overall system state tree was constructed and evaluated, and this procedure will be discussed in the following sections.

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Table 3.5. Initiating events and frequencies utilized in PTS study

Initiating Event	Assumed Frequency (/RY)
Reactor/Turbine Trip	6.0E+00
Excessive MFW	1.0E-01
LSLB	1.0E-03
SSLB	1.0E-02
Loss of MFW	5.0E-01
SB LOCA1	1.0E-01
SB LOCA2	1.0E-02
Inadvertent SI	1.0E-02
SGTR	1.0E-02

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3.6.1 Reactor/Turbine Trip

3.6.1.1 Initial Conditions and Conditioning Event Assumptions

The initial conditions for this initiator assumed that reactor power is greater than 45% and that the decay heat capability of the core was approximately the ANS standard. Based on the specified initial conditions, the following conditioning event assumptions were made:

1. Following the trip, the steam line pressure in both steam lines will be sufficient to challenge both the turbine bypass system and the secondary-side relief valves (at least momentarily). Thus, the high SL pressure branches for both steam lines on the SPC system state tree are activated.

2. The MFW system is operating with both MFW pumps on-line, and in the absence of any secondary side pressure control difficulty, the MFW line pressure will remain above the EFW initiation set point. If any secondary pressure-control valve fails to reseat (either a TBV or a SSRV), it is assumed that the MFW line pressure will decrease below the EFW set point (725 psi). Further, should a secondary side depressurization take place, it is assumed that such a depressurization will not be rapid enough to induce a false high-level signal from the instruments. Therefore, it is assumed that the SG level signals are nominally correct for all cases.

3. With regard to the ECC system, it is assumed that, if no secondary-side depressurization or overfeed occurs, no demand for an ECCS response will be made. If either a depressurization or an overfeed takes place, it is further assumed that only the HPI system will be demanded.

4. It is assumed that the PORV block valve is closed 50% of the time. This assumption is conservative in that it forces the safety valves to open more frequently. The safety valves cannot have block valves, and thus a failed-open safety valve is a nonisolable break.

Since this is the first initiator to be discussed, it is useful to observe the effect of these conditioning event assumptions on the four system state trees. The four trees are given by Figures 3.17 through 3.20. As indicated in assumption 2 above, two FW system state trees result from the depressurization/no-depressurization cases.

3.6.1.2 Reactor/Turbine Trip System State Tree

Combining the four system state trees given by Figures 3.17 through 3.20 results in an overall reactor trip system state tree comprised of 448 end-states with frequencies greater than 1.0E-06. Additionally, the end states eliminated in the pruning process represent an additional frequency of 1.9E-04. As an alternate perspective, the surviving end-states have been grouped according to the system (or systems) that malfunctioned during the transient. These results are presented in Table 3.6. As expected, the case of no malfunctions clearly dominates in terms of frequency of occurrence.

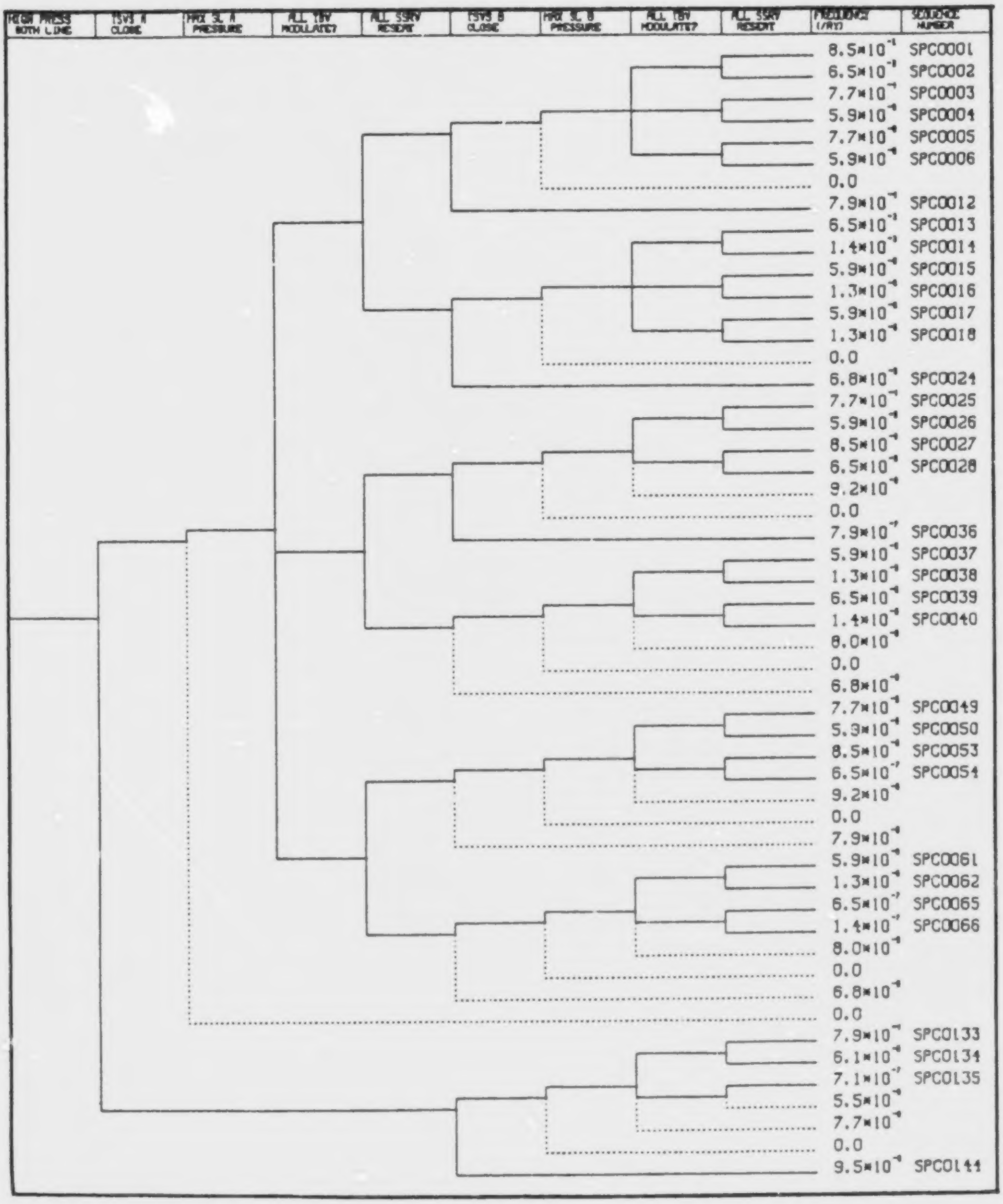
HIGH PRESS BOTH LINE
TSVS A CLOSE
MAX SL A PRESSURE
ALL TBV MODULATE?
ALL SSRV RESEAT
TSVS B CLOSE
MAX SL B PRESSURE
ALL TBV MODULATE?
ALL SSRV RESEAT
FREQUENCY (/RY)
SEQUENCE NUMBER
8.5×10^{-1} SPC0001
6.5×10^{-3} SPC0002
7.7×10^{-7} SPC0003
5.9×10^{-4} SPC0004
7.7×10^{-4} SPC0005
5.9×10^{-4} SPC0006
0.0
7.9×10^{-7} SPC0012
6.5×10^{-3} SPC0013
1.4×10^{-3} SPC0014
5.9×10^{-4} SPC0015
1.3×10^{-4} SPC0016
5.9×10^{-4} SPC0017
1.3×10^{-4} SPC0018
0.0
6.8×10^{-4} SPC0024
7.7×10^{-7} SPC0025
5.9×10^{-4} SPC0026
8.5×10^{-4} SPC0027
6.5×10^{-4} SPC0028
9.2×10^{-4}
0.0
7.9×10^{-7} SPC0036
5.9×10^{-4} SPC0037
1.3×10^{-4} SPC0038
6.5×10^{-4} SPC0039
1.4×10^{-4} SPC0040
8.0×10^{-4}
0.0
6.8×10^{-4}
7.7×10^{-4} SPC0049
5.3×10^{-4} SPC0050
8.5×10^{-4} SPC0053
6.5×10^{-7} SPC0054
9.2×10^{-4}
0.0
7.9×10^{-4}
5.9×10^{-4} SPC0061
1.3×10^{-4} SPC0062
6.5×10^{-7} SPC0065
1.4×10^{-7} SPC0066
8.0×10^{-4}
0.0
6.8×10^{-4}
0.0
7.9×10^{-7} SPC0133
6.1×10^{-4} SPC0134
7.1×10^{-7} SPC0135
5.5×10^{-4}
7.7×10^{-4}
0.0
9.5×10^{-4} SPC0144

Figure 3.17 Secondary pressure control system state tree; initiator: unspecified.

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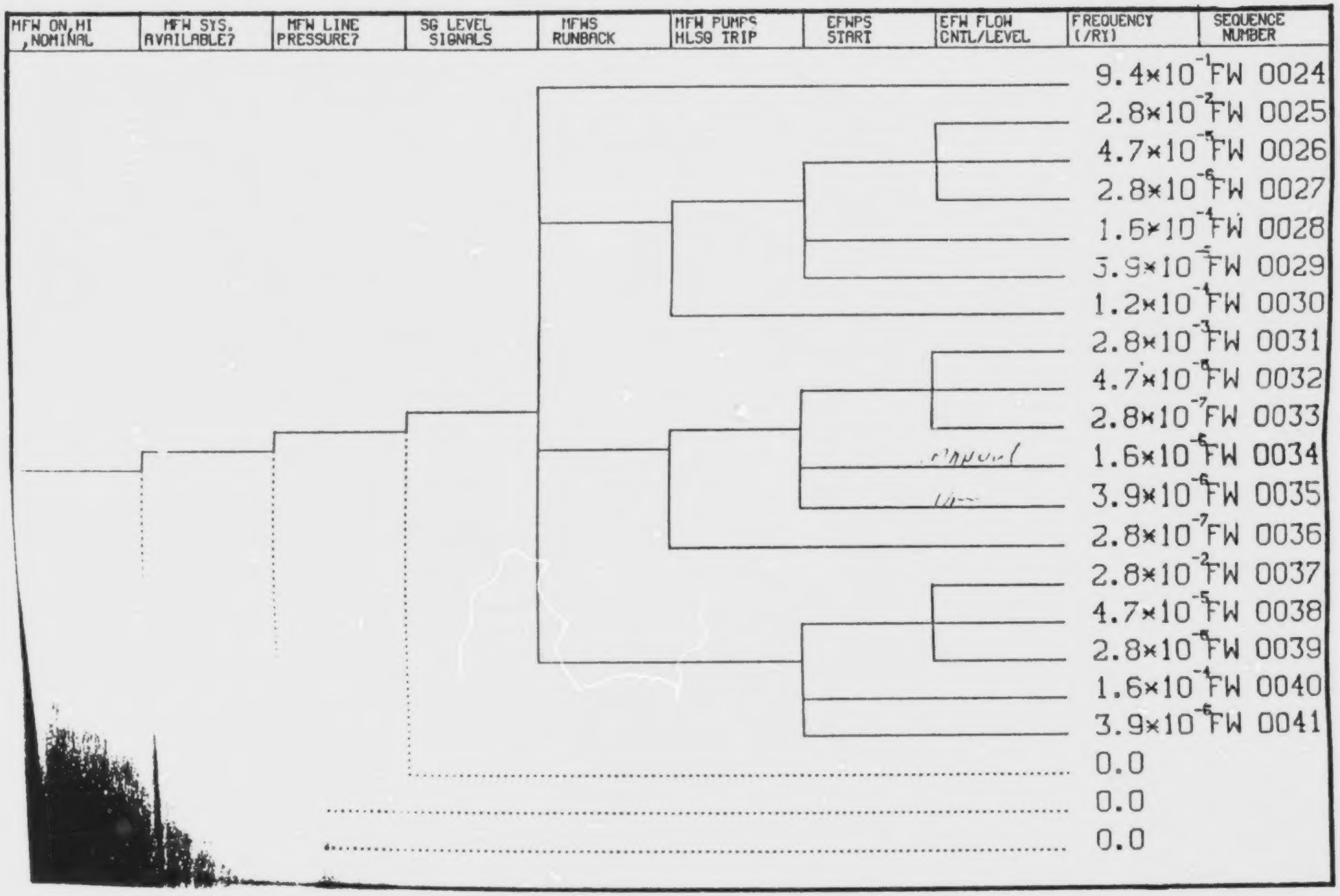
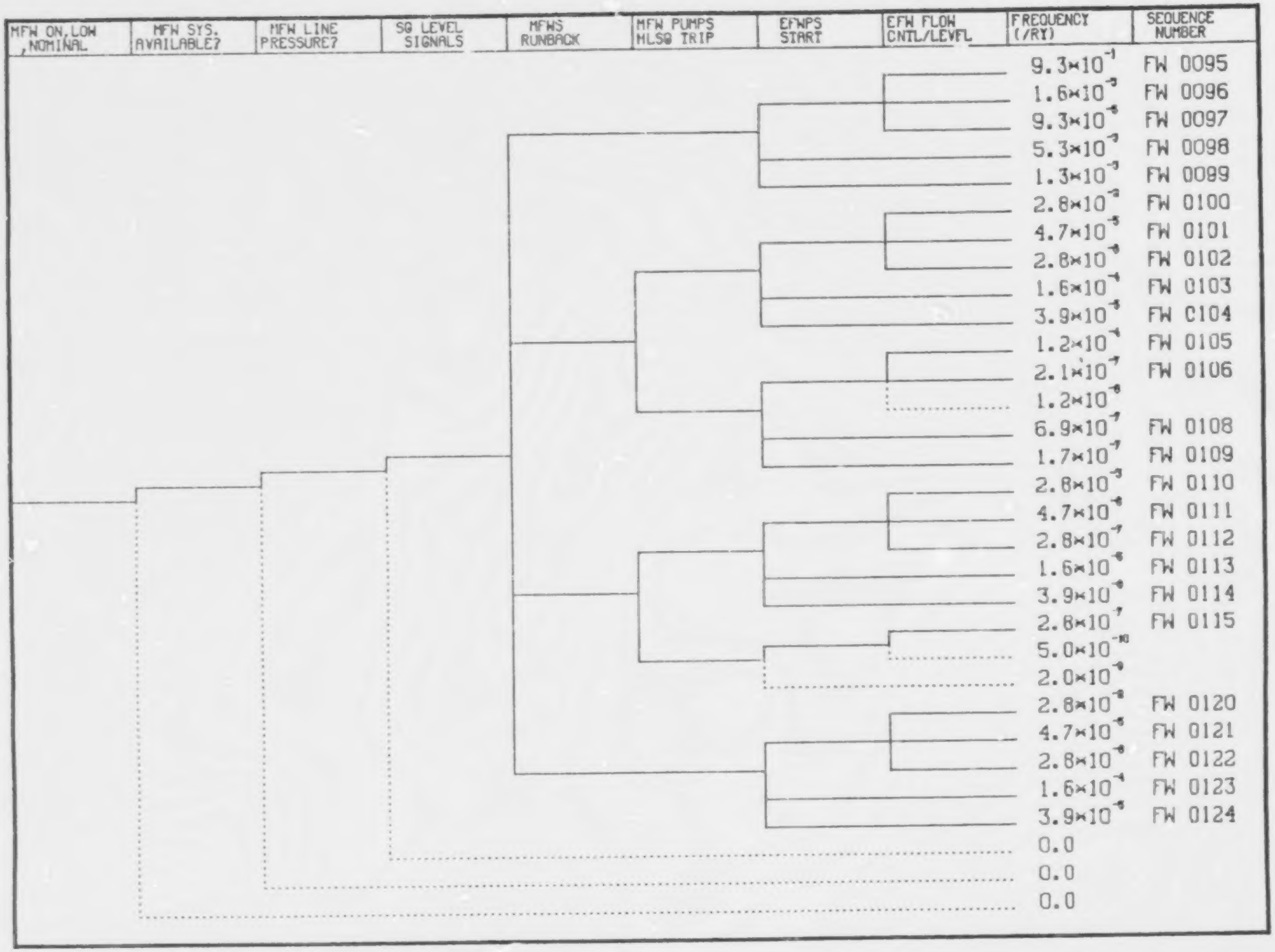

MFW ON,HI ,NOMINAL
MFW SYS. AVAILABLE?
MFW LINE PRESSURE?
SG LEVEL SIGNALS
MFWS RUNBACK
MFW PUMPS HLSG TRIP
EFWPS START
EFW FLOW CNTL/LEVEL
FREQUENCY (/RY)
SEQUENCE NUMBER
9.4×10⁻¹ FW 0024
2.8×10⁻² FW 0025
4.7×10⁻⁵ FW 0026
2.8×10⁻⁶ FW 0027
1.6×10⁻⁴ FW 0028
3.9×10⁻⁵ FW 0029
1.2×10⁻⁴ FW 0030
2.8×10⁻³ FW 0031
4.7×10⁻⁶ FW 0032
2.8×10⁻⁷ FW 0033
1.6×10⁻⁵ FW 0034
3.9×10⁻⁶ FW 0035
2.8×10⁻⁷ FW 0036
2.8×10⁻² FW 0037
4.7×10⁻⁵ FW 0038
2.8×10⁻⁶ FW 0039
1.6×10⁻⁴ FW 0040
3.9×10⁻⁶ FW 0041
0.0
0.0
0.0
manual

Figure 3.18a Feedwater system state tree; initiator: unspecified.



MFW ON,LOW ,NOMINAL
MFW SYS. AVAILABLE?
MFW LINE PRESSURE?
SG LEVEL SIGNALS
MFWS RUNBACK
MFW PUMPS HLSG TRIP
EFWPS START
EFW FLOW CNTL/LEVEL
FREQUENCY (/RY)
SEQUENCE NUMBER
9.3×10^{-1} FW 0095
1.6×10^{-3} FW 0096
9.3×10^{-6} FW 0097
5.3×10^{-3} FW 0098
1.3×10^{-3} FW 0099
2.8×10^{-2} FW 0100
4.7×10^{-5} FW 0101
2.8×10^{-6} FW 0102
1.6×10^{-4} FW 0103
3.9×10^{-5} FW C104
1.2×10^{-4} FW 0105
2.1×10^{-7} FW 0106
1.2×10^{-8}
6.9×10^{-7} FW 0108
1.7×10^{-7} FW 0109
2.8×10^{-3} FW 0110
4.7×10^{-6} FW 0111
2.8×10^{-7} FW 0112
1.6×10^{-5} FW 0113
3.9×10^{-6} FW 0114
2.8×10^{-7} FW 0115
5.0×10^{-10}
2.0×10^{-9}
2.8×10^{-3} FW 0120
4.7×10^{-5} FW 0121
2.8×10^{-6} FW 0122
1.6×10^{-4} FW 0123
3.9×10^{-5} FW 0124
0.0
0.0
0.0

Figure 3.18b Feedwater system state tree; initiator: unspecified.

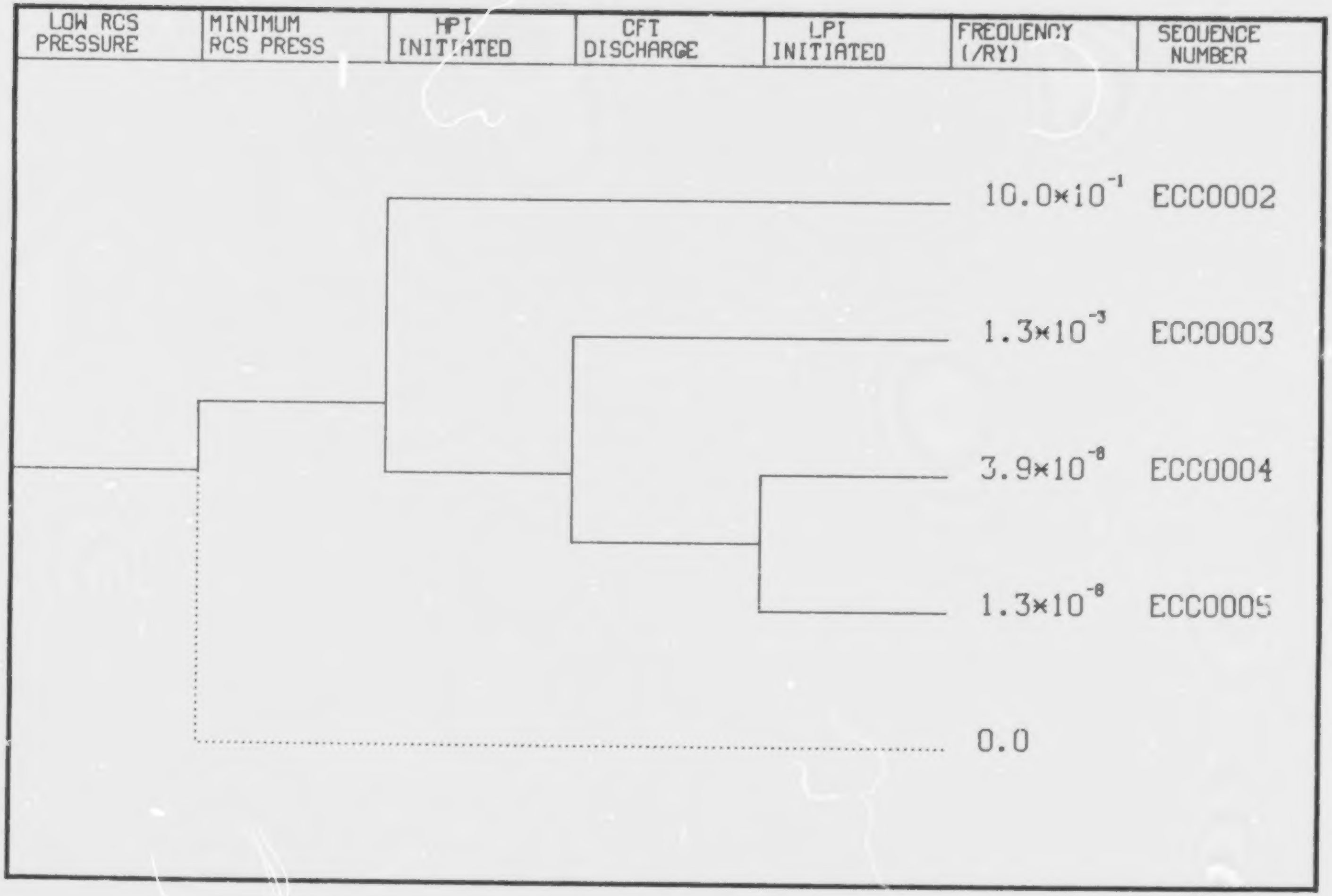

LOW RCS PRESSURE
MINIMUM RCS PRESS
HPI INITIATED
CFT DISCHARGE
LPI INITIATED
FREQUENCY (/RY)
SEQUENCE NUMBER
10.0×10^-1
ECC0002
1.3×10^-3
ECC0003
3.9×10^-8
ECC0004
1.3×10^-8
ECC0005
0.0

Figure 3.19 Primary loop ECCS system state tree; initiator: low RCS pressure.

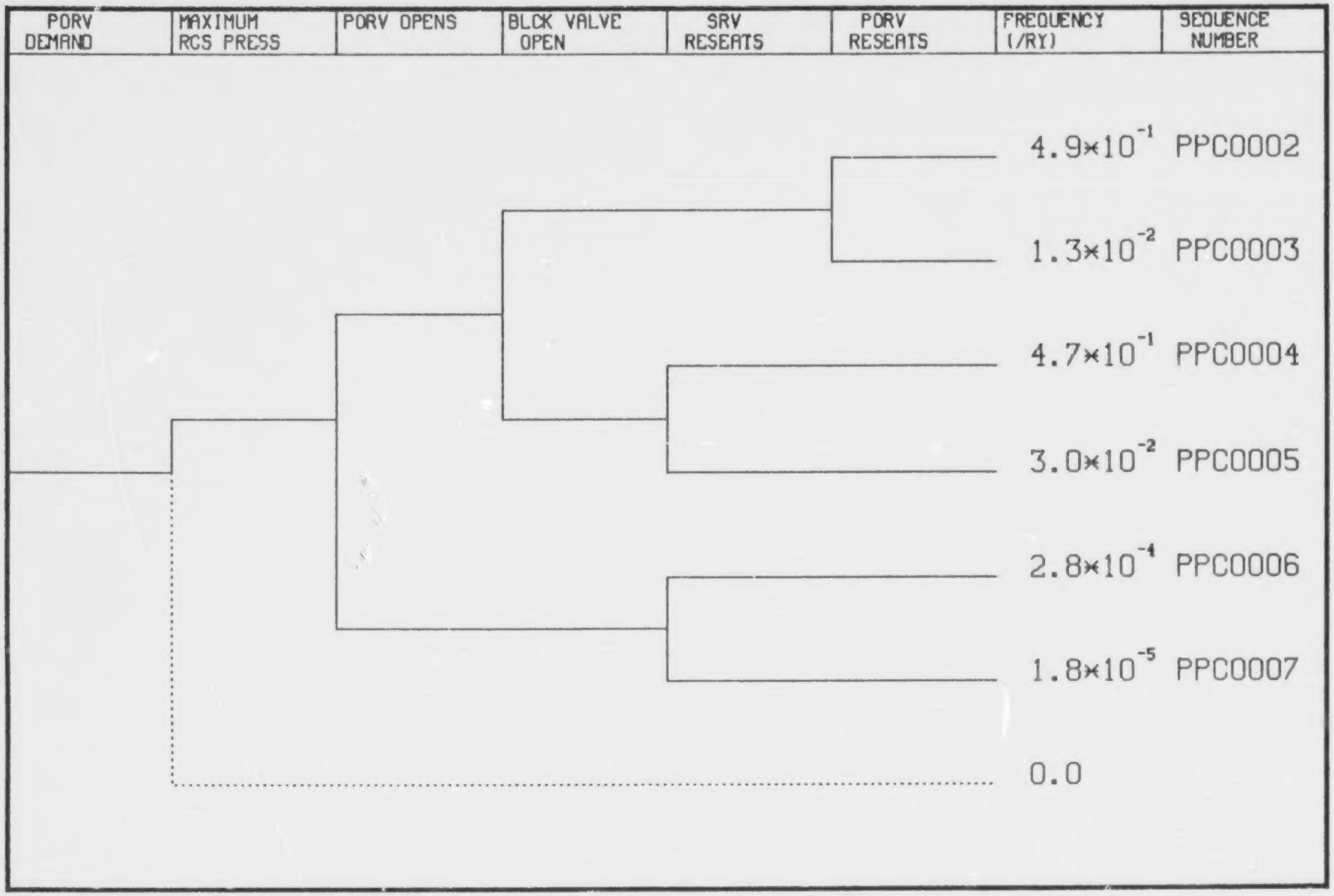

PORV DEMAND
MAXIMUM RCS PRESS
PORV OPENS
BLCK VALVE OPEN
SRV RESEATS
PORV RESEATS
FREQUENCY (/RY)
SEQUENCE NUMBER
4.9×10^-1 PPC0002
1.3×10^-2 PPC0003
4.7×10^-1 PPC0004
3.0×10^-2 PPC0005
2.8×10^-4 PPC0006
1.8×10^-5 PPC0007
0.0

Figure 3.20 Primary pressure control system state tree.

Table 3.6 Reactor trip transient summary

System	Number of Transients	Frequency (/RY)
None	12	5.0+0
SPC	79	7.8-1
FWC	43	1.6-1
SPC + FWS	114	3.0-2
SPC + ECC	13	1.0-3
SPC + PPC	73	7.0-2
FWS + ECC	--	--
FWS + PPC	27	2.7-4
SPC + FWS + ECC	9	3.5-5
SPC + FWS + PPC	70	2.8-3
SPC + ECC + PPC	8	8.6-5
	448	6.0+0
Residual		1.9-4

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In terms of number of end-states, the transients involving malfunctions of both the secondary pressure control and feedwater systems are the most numerous. Further, the transients which involve only a SPC malfunction are the most likely of those transients that involve malfunctions. A complete list of all transients for this initiator is given in Appendix A, Table A.1 together with the classification of each transient into an appropriate category.

3.6.2 Excessive MFW: Initial Conditions and Conditioning Event Assumptions

The initial conditions assumed for this initiator are reactor power greater than 45% and the ANS standard decay heat. Based on the initial conditions, the following conditioning event assumptions were made:

1. Eventual reactor trip on either low RCS pressure or unbalanced primary thermal-hydraulic parameters.
2. Following the reactor trip, assumption that the pressure in both steam lines is sufficient to challenge the secondary side relief valves.
3. MFW system operating with both MFW pumps on line. In the absence of a secondary side depressurization due to a stuck-open valve, it is assumed that the MFW line pressure remains above the EFW initiation setpoint.
4. The primary-side pressure, assumed to decrease (due to the MFW overfeed) to below the HPI set point, but to remain above the CFT set point.

5. Assumption that the PORV block valve is closed 50% of the time.

3.6.3 Large Steam Line Break (LSLB): Initial Conditions and Conditioning Event Assumptions

The assumed initial conditions are reactor power greater than 45% power and ANS standard decay heat. The size of the large steam line break is assumed to be sufficient to depressurize the steam generator to below the EFW initiation point. Based on these initial conditions, the following conditioning events were selected:

1. Low SL pressure in the broken steam line to the extent that neither the relief valves nor the TBVs are challenged. It is assumed that the pressure in the unaffected steam line following the reactor trip is sufficient to challenge both the secondary relief valves and the turbine bypass valves.

2. MFW system is on, but MFW line pressure decreases to below the EFW initiation point.

3. Minimum primary system pressure between 610 and 1500 psi.

4. PORV block valve closed 50% of the time.

3.6.4 Small Steam Line Break

Initial Conditions and Conditioning Event Assumptions

The initial conditions assumed are reactor power greater than 45% and ANS standard decay heat. The size of the steam line break is assumed to be insufficient to depressurize the steam generator to the EFW initiation point. Given these assumptions, the following conditioning event assumptions were made:

1. Steam line pressure in the affected steam line is low enough not to challenge either SSRVs or TBVs. Pressure in the unaffected steam line is sufficient (following reactor trip) to challenge both the SSRVs and the TBVs.

2. MFW system is initially on line; the depressurization caused by the steam line break is not sufficient to initiate the EFW.

3. Minimum RCS pressure reached is between 610 and 1500 psi (only HPI system demanded).

4. Maximum RCS pressure is sufficient to challenge both the PORV and the SRVs.

5. PORV block valve is closed 50% of the time.

3.6.5 Loss of MFW (LOMFW): Initial Conditions and Conditioning Event Assumptions

The initial conditions for the LOMFW initiator are reactor power greater than 45% and the ANS standard decay heat. A trip of the MFW pumps at the start of the transient is assumed. Based on the initial conditions, the following conditioning event assumptions were made:

1. Steam line pressure in both steam lines (following reactor trip) sufficient to challenge both the relief valves and turbine bypass system.
2. MFW system is unavailable.
3. Primary-side pressure is assumed to be sufficient (given a failure of the EFW system to initiate) to challenge both the PORV and the code safety valves.
4. No ECCS demand is assumed if a secondary-side overfeed does not occur. Given an overfeed, it is assumed that only the HPI system is demanded.
5. The PORV block valve is closed 50% of the time.

3.6.6 Small Break LOCA 1: Initial Conditions and Conditioning Event Assumptions

The initial conditions assumed for this initiator are reactor power greater than 45% and ANS standard decay heat. The size of the break is assumed to be large enough to depressurize the primary system below the HPI set-point.

Additionally, it is assumed the break is small enough that the HPI system is capable of repressurizing to the PORV/SRV set-points. Based on these specifications, the following conditioning event assumptions were made:

1. Reactor trip on low RCS pressure.
2. High steam line pressure in both steam lines following the reactor trip. Both SSRVs and TBVs are challenged.
3. MFW system operating. MFW line pressure greater than 725 psi for cases with no secondary pressure control malfunctions; less than 725 psi for any secondary pressure control malfunction (TBVs or SSRVs).
4. Only the HPI system will be demanded.
5. PORV block valve closed 50% of the time.

3.6.7 Small Break LOCA 2: Initial Conditions and Conditioning Event Assumptions

The initial conditions assumed for this initiator are the same as for the small-break LOCA 1 case. However, the size of this break is assumed to be large enough to prevent the HPI system from repressurizing the primary to either the PORV or SRV set points. The following conditioning event assumptions were made:

1. Reactor trip on low RCS pressure.
2. High steam line pressure in both steam lines following the reactor trip challenging both the TBVs and the SSRVs.

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3. MFW system operating. MFW line pressure remains greater than 725 psi if no secondary depressurization occurs.

4. Only the HPI system is demanded.

3.6.8 Inadvertent Safety Injection (SI): Initial Conditions and Conditioning Event Assumptions

The initial conditions assumed are reactor power greater than 45% and ANS standard decay heat. The following conditioning event assumptions were made:

1. Reactor trip on high RCS pressure.

2. Maximum steam line pressure in both steam lines. Both TBVs and SSRVs are challenged.

3. MFW system operating. MFW line pressure greater than 725 psi for no secondary-side depressurization.

4. Maximum primary-system pressure--both PORV and SRVs challenged.

5. PORV block valve closed 50% of the time.

3.6.9 Categorization of Initiator-Specific Asymptotic Transients

Table A.1 (Appendix A) summarizes the relevant information for the 1013 initiator-specific asymptotic end-states that survive the pruning process. In particular, each transient is described by a sequence number (which is the position of the particular transient in the set of 6,935,760 possible transients) and four system state branch descriptors that delineate the particular system state tree end-states reached for the transient. Additionally, the estimated asymptotic event frequencies are tabulated. Finally, the system state categories appropriate for each transient (based on the particular system state end-states) are listed. Examination of all the transients will reveal that certain category groupings occur more than once. Hence, in order to reduce the number of transients to a more manageable level, transients with identical categorizations were combined. The transients thus obtained are given in Table A.2.

3.7 Effect of Operator Actions

As discussed previously, it is necessary to construct an operator action tree detailing those operator actions that significantly affect the thermal-hydraulic behavior of the system for each unique end-state. However, in keeping with the categorization of the end-states, it is assumed that an operator action tree constructed for a representative transient in the category can be applied to all transients in the category.

As noted in the discussion of conditioning events, it is assumed that any secondary-side depressurization or overfeed will lead to a demand for the HPI system because of low primary system pressure. Thus, for SPC Category 1 transients, the first relevant operator action is the trip of the reactor coolant pumps on the valid ECCS demand signal. This action is particularly important on B&W plants because the trip of the RC pumps permits the vent valves to function in a manner to mitigate the cooldown of the vessel. On diagnosing the problem as a secondary-side rupture or excessive steam demand, the next relevant step is the isolation of both SGs as called for

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in the procedure. Once the broken secondary loop has been identified, the next action is the restoration of feedwater to the unaffected loop (and hence the restoration of the secondary heat sink). The remaining relevant operator actions consist of throttling the HPI system and the restart of one RC pump per loop when a $50^{o}F$ subcooling margin has been attained.

Since a detailed task analysis of the relevant procedure was not performed, generic human error probabilities (HEPs) were used to estimate the mitigative potential of the generator. For most actions a HEP value of 1.0 x 10^{-2} was employed for the first error in a sequence. The HEP for each succeeding error was obtained by doubling the previous value. For certain specific actions for which extensive training had been accomplished (such as the trip of the RC pumps on a valid ECC signal), a HEP value of 5.0 x 10^{-3} was employed. It should be noted that each operator action requires that a time interval be associated with it. The generic probabilities discussed above are based on operator action within a 10-min time frame except for the RC pump trip in which it is assumed that a 30-s time frame was sufficient for operator action, if it was to be taken at all.

The operator action tree, constructed using the above assumptions, is depicted by Figure 3.21 and is strictly appropriate to only Category (1,0,0,0) transients. However, it is observed that PSC02 transients differ only in the fact that the break can be isolated (e.g., a stuck-open TBV versus a SSRV). It should be noted, however, that a successful isolation of SGs will result in similar conditions in both steam lines (due to the closure of the turbine bypass valves), which will make the diagnosis of

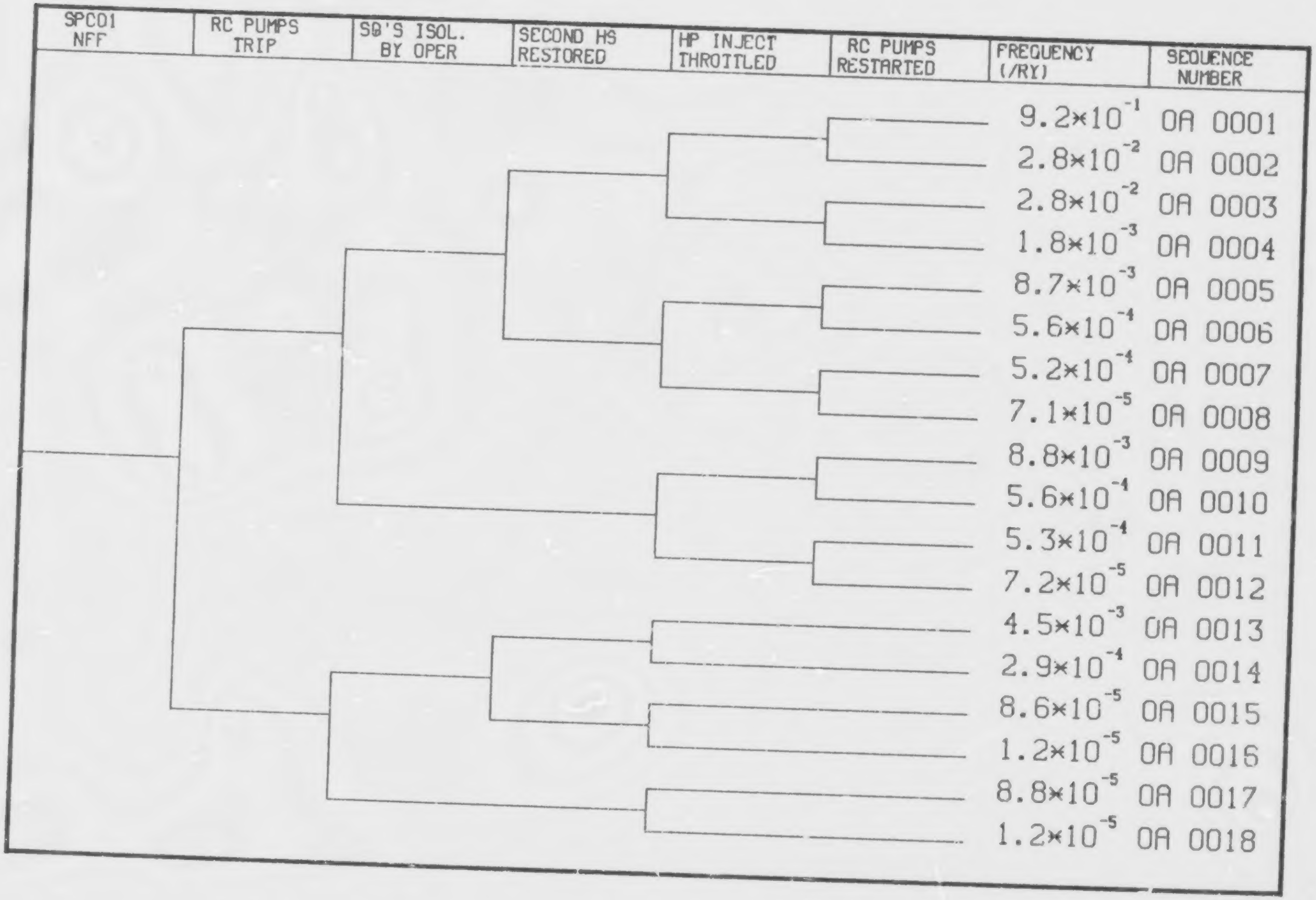
SPC01 NFF
RC PUMPS TRIP
SG'S ISOL. BY OPER
SECOND HS RESTORED
HP INJECT THROTTLED
RC PUMPS RESTARTED
FREQUENCY (/RY)
SEQUENCE NUMBER
9.2×10^{-1} OA 0001
2.8×10^{-2} OA 0002
2.8×10^{-2} OA 0003
1.8×10^{-3} OA 0004
8.7×10^{-3} OA 0005
5.6×10^{-4} OA 0006
5.2×10^{-4} OA 0007
7.1×10^{-5} OA 0008
8.8×10^{-3} OA 0009
5.6×10^{-4} OA 0010
5.3×10^{-4} OA 0011
7.2×10^{-5} OA 0012
4.5×10^{-3} OA 0013
2.9×10^{-4} OA 0014
8.6×10^{-5} OA 0015
1.2×10^{-5} OA 0016
8.8×10^{-5} OA 0017
1.2×10^{-5} OA 0018

Figure 3.21 SPC01 operator action tree; initiator: SPC01.

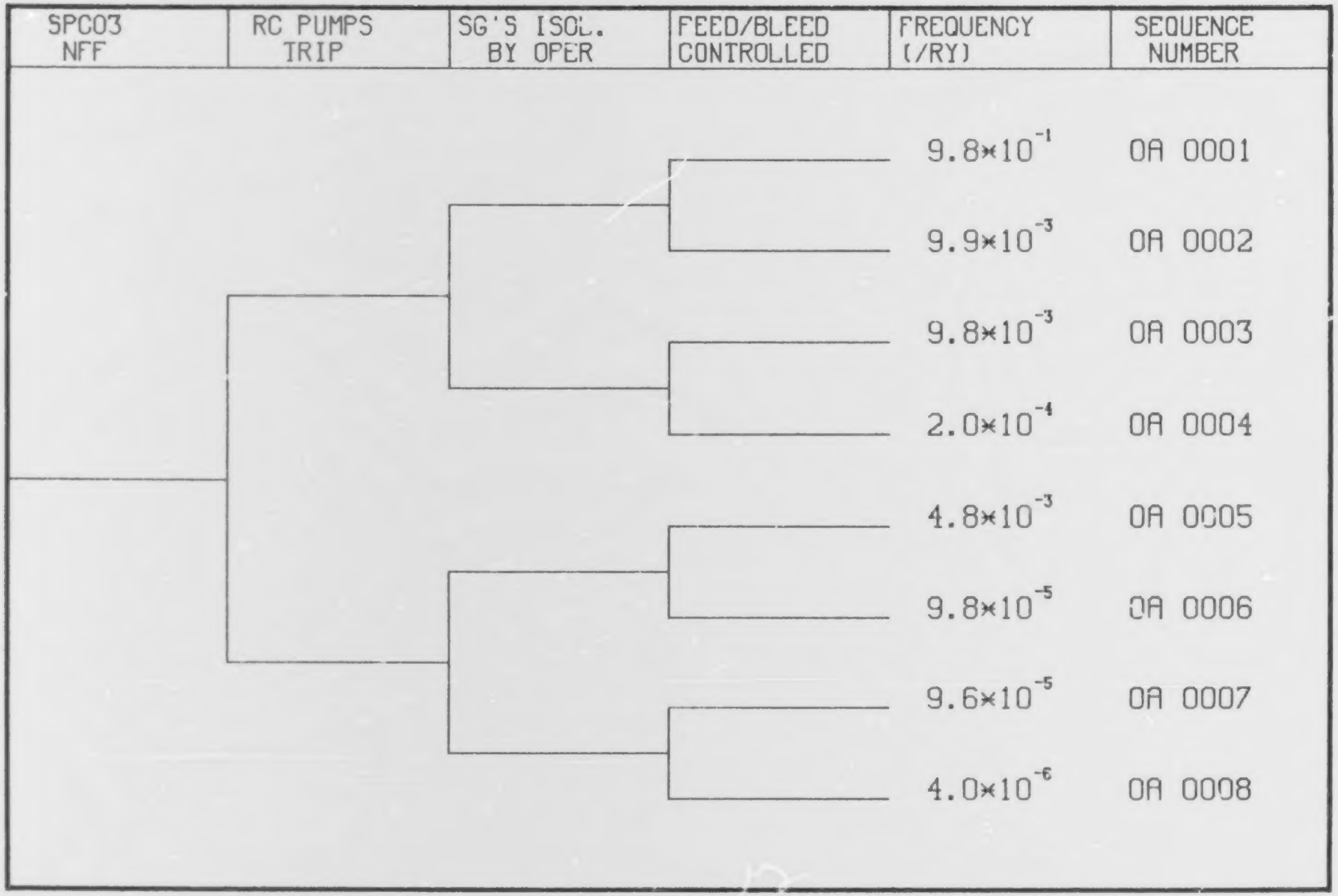
SPC03 NFF
RC PUMPS TRIP
SG'S ISOL. BY OPER
FEED/BLEED CONTROLLED
FREQUENCY (/RY)
SEQUENCE NUMBER
9.8×10^-1
OA 0001
9.9×10^-3
OA 0002
9.8×10^-3
OA 0003
2.0×10^-4
OA 0004
4.8×10^-3
OA 0005
9.8×10^-5
OA 0006
9.6×10^-5
OA 0007
4.0×10^-6
OA 0008

Figure 3.22 SPC03 operator action tree; initiator: SPC03.

the affected steam line more difficult. Attempted restoration of the secondary heat sink, however, will indicate the affected steam line.

For an SPC Category 3 transient, isolation of the steam generators will not isolate the break, resulting in a depressurization of both SGs. Because neither GS can be restored as a heat sink, it is assumed the reactor operators will provide cooling to the core via the HPI system (i.e., a feed/bleed mode). The operator action tree for this case is given by Figure 3.22.

For SPC Category 4, one steam generator will appear to be unaffected. However, attempted restoration will indicate that one or more TBVs are stuck open. It is assumed that the steam generator can be restored as a viable secondary heat sink by limiting the steam flow through the turbine bypass block valves. Thus, the appropriate operator action tree for the case is given by Figure 3.22. In an analogous fashion, this same operator action tree can also be used for SPC Category 5 transients.

As noted above, SPC Category 6 transients principally account for failures of the TSVs to close on both lines. If the turbine governing valves close, these transients can be categorized as SPC Category 0. However, a failure to trip the turbine, given a reactor trip, will result in an excessive steam demand relative to the power output of the reactor. Assuming a turbine trip failure probability of 2×10^{-4}/demand yields a revised frequency for SPC Category 6 transients of 1.2×10^{-3}. In this case, manual tripping of the turbine, or failing that, manual closure of the turbine stop valves and/or the turbine governing valves will effectively isolate the break. It is assumed that, should the manual isolation of the turbine not be accomplished, the transient will be treated as a "double" steam line rupture (SPC Category 3) with similar operator actions. The operator action tree for this case is given by Figure 3.23.

For those actions involving only malfunctions of the feedwater system [Category (0,N,0,0)], a different set of operator action trees is appropriate. The first nontrivial Category (0,1,0,0) involves unsustained MFW overfeeds to one or both of the SGs. It is assumed that the time frame of this overfeed is short enough to preclude any operator action before the high-level SG trip system terminates the transient itself.

For FWS Categories 2 and 3, which involve sustained overfeeds by the MFW and EFW systems respectively, the operator action tree given by Figure 3.24 was constructed. An implicit assumption inherent in his tree is the initiation of he HPI system on low RCS pressure, thereby requiring a trip of the RC pumps. The FWS Category, however, is characterized by a very different thermal-hydraulic behavior. Since this category represents a loss of both the MW and EFW systems on reactor trips, an automatic HPI initiation on low RCS pressure is not expected, and the RC pump trip will not be procedurally required. Additionally, one reason for the non-initiation of the EFW system is that the system was originally in a manual mode of operation, requiring only a switch back to automatic mode to effect EFW supply. The HEP assigned to this branch was 0.01. Alternatively the EFW system may be unavailable for other reasons. However, because of the interconnections possible between the EFW system of the three Oconee units, a 0.98 probability of success of EFW restoration was assumed for this branch. Also, a potential operator error of commission, restoration

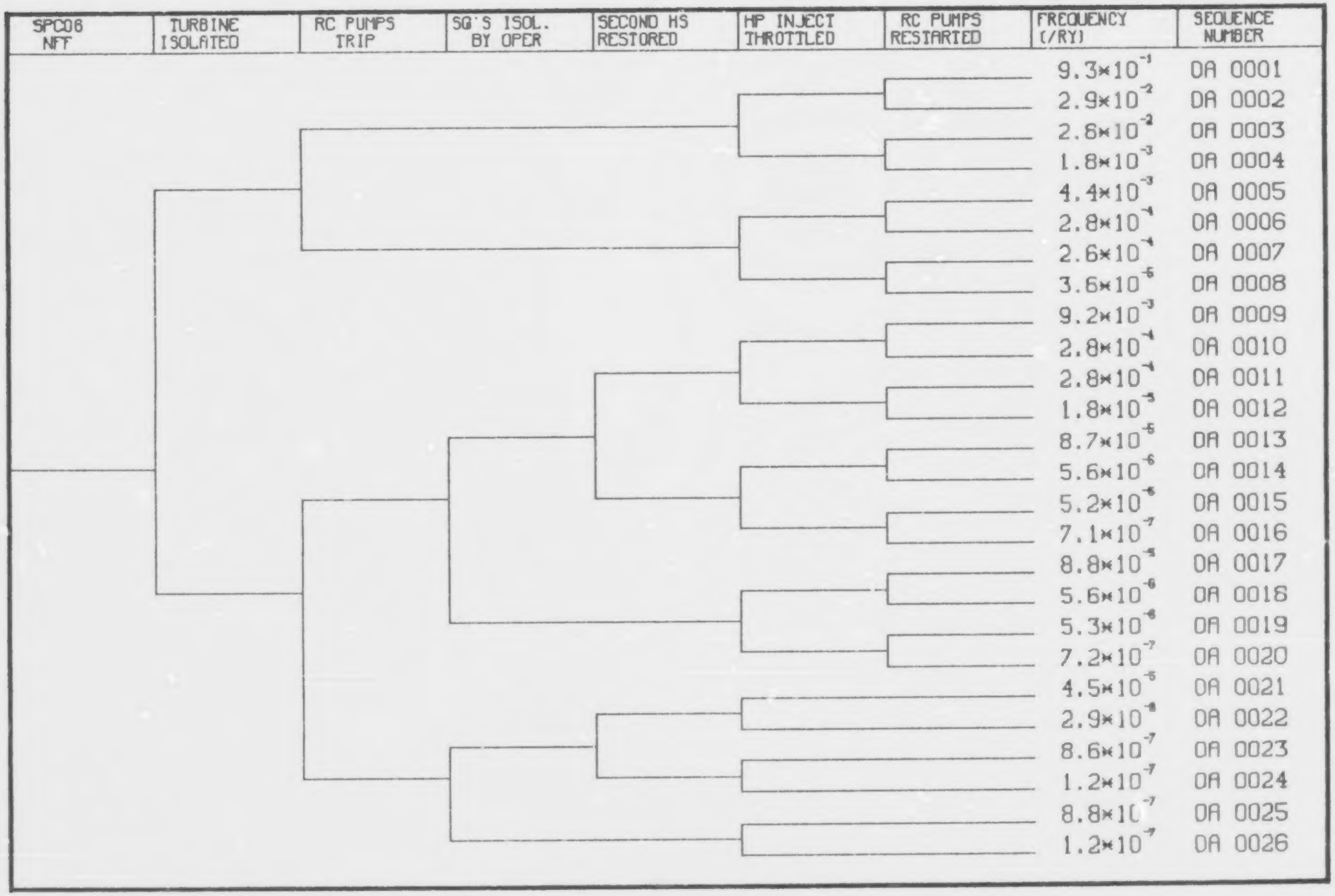
SPC06 NFT
TURBINE ISOLATED
RC PUMPS TRIP
SG'S ISOL. BY OPER
SECOND HS RESTORED
HP INJECT THROTTLED
RC PUMPS RESTARTED
FREQUENCY (/RY)
SEQUENCE NUMBER
9.3×10^{-1} OA 0001
2.9×10^{-2} OA 0002
2.8×10^{-2} OA 0003
1.8×10^{-3} OA 0004
4.4×10^{-3} OA 0005
2.8×10^{-4} OA 0006
2.6×10^{-4} OA 0007
3.6×10^{-6} OA 0008
9.2×10^{-3} OA 0009
2.8×10^{-4} OA 0010
2.8×10^{-4} OA 0011
1.8×10^{-5} OA 0012
8.7×10^{-5} OA 0013
5.6×10^{-6} OA 0014
5.2×10^{-6} OA 0015
7.1×10^{-7} OA 0016
8.8×10^{-5} OA 0017
5.6×10^{-6} OA 0018
5.3×10^{-6} OA 0019
7.2×10^{-7} OA 0020
4.5×10^{-5} OA 0021
2.9×10^{-6} OA 0022
8.6×10^{-7} OA 0023
1.2×10^{-7} OA 0024
8.8×10^{-7} OA 0025
1.2×10^{-7} OA 0026

Figure 3.23 SPC06 operator action tree; initiator: SPC06.

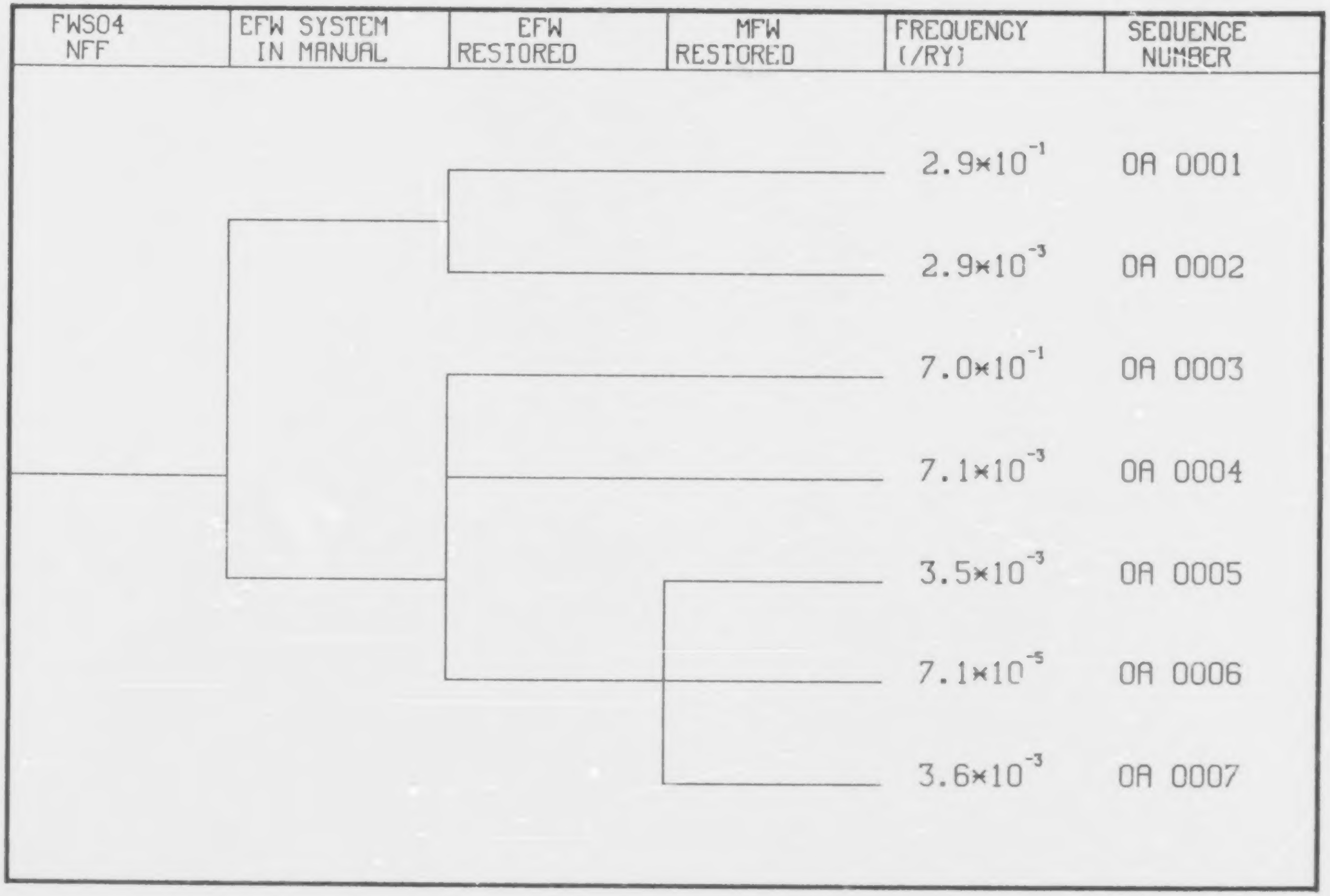
FWS04
NFF
EFW SYSTEM
IN MANUAL
EFW
RESTORED
MFW
RESTORED
FREQUENCY
(/RY)
SEQUENCE
NUMBER
2.9×10^-1
OA 0001
2.9×10^-3
OA 0002
7.0×10^-1
OA 0003
7.1×10^-3
OA 0004
3.5×10^-3
OA 0005
7.1×10^-5
OA 0006
3.6×10^-3
OA 0007

Figure 3.24 FWS04 operator action tree; initiator: FWS04.

of EFW in an uncontrolled manner (ERW overfeed), was also included with a HEP of 0.01. Should EFW restoration not be possible, a probability of 0.5 of MFW restoration in an uncontrolled manner was employed.

For most of the transients involving both SPC and FWS malfunctions, isolation of the steam generators mitigates the FWS malfunction. Therefore, the appropriate operator action trees for categories designated (N,M,O,O) are the same trees as discussed above for Categories (N,O,O,O). It should be emphasized, however, that a SPC malfunction coupled with an overfeed results in more severe thermal-hydraulic behavior even though the operator actions necessary to mitigate the transient remain the same. For those combined transients involving FWS Category 4 (loss of all FW), however, a modified form of the SPC operator action trees is employed. The modification consists of changing the probability of heat sink restoration to agree with that indicated by Figure 3.24 to account for the possibility of uncontrolled EFW or MFW restoration (1.0E-02 and 7.1E-05, respectively).

For those transients involving a failure of the HPI to initiate, a conservative approximation is made. Because the HPI system will exacerbate a PTS transient (higher pressure and lower temperature in the downcomer), those transients that involve HPI failure are included in the category representing HPI success.

For those transients in which the pressure in the RC system reaches the PORV/SRV set points, the operator actions necessary to deal with a potential LOCA situation must be addressed. Two types of potential LOCAs are represented by the categorization process--LOCAs isolable via the PORV block valve and non-isolable LOCAs involving the SRVs. It should be noted that the LOCA situation may not be initially apparent in many of the transients because the effects may be masked by the overcooling of the primary system. Hence in this study, the conservative assumption was made that for those transients involving secondary-side depressurization or excessive feedwater, the LOCA situation will only be dealt with following correction of the secondary side malfunction. For PORV LOCAs, the system can be stabilized by closing the PORV block valve and throttling back the HPI, given that the secondary heat sink has been restored. Thus, an additional operator action, closing the PORV block valve, should be incorporated into the operator action tree. For the nonisolable LOCA, the ECCS flow rate will match the break flow at a pressure lower than operating pressure. In both cases, however, the appropriate operator action depends on the availability of the secondary heat sink. If it is available, it is assumed that no operator action is taken to alter the feed/bleed mode in the RCS. If the SGs are not (or cannot be) restored, an operator action to depressurize the primary system to utilize the LPI system is appropriate and is included on those branches for which secondary side cooling is not restored. Figure 3.25 depicts the operator action tree for Category (1,0,0,1) as an example.

The above discussion is intended to serve as background for the discussion of initiator-specific transients. In many cases, the operator action trees postulated are directly applicable to the transients involved. In certain cases, however, the initiators themselves require modification of the generic operator action trees postulated based on the transient categorization. The necessary modifications will be indicated in the discussion of each specific initiator where appropriate.

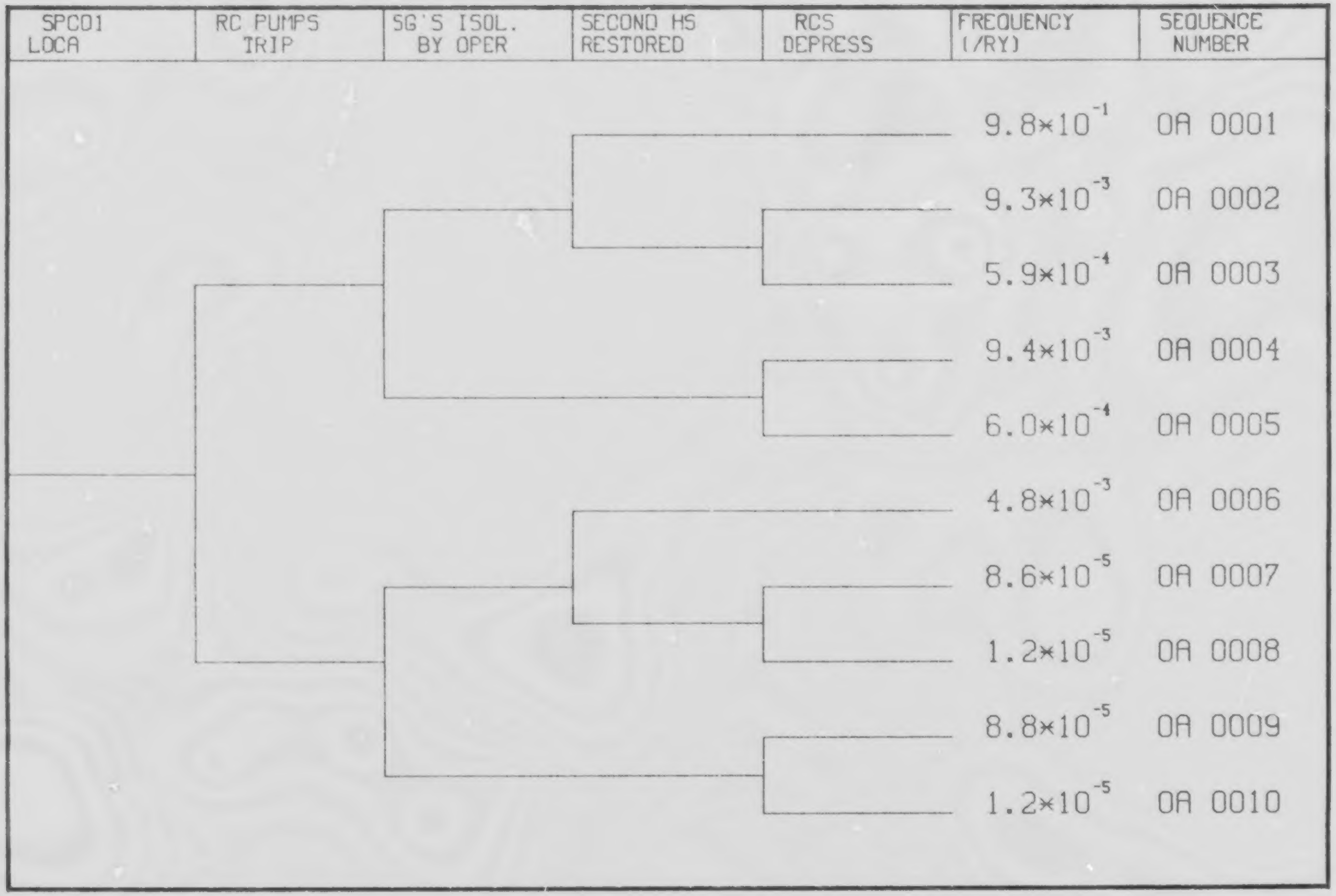
SPC01 LOCA
RC PUMPS TRIP
SG'S ISOL. BY OPER
SECOND HS RESTORED
RCS DEPRESS
FREQUENCY (/RY)
SEQUENCE NUMBER
9.8×10⁻¹ OA 0001
9.3×10⁻³ OA 0002
5.9×10⁻⁴ OA 0003
9.4×10⁻³ OA 0004
6.0×10⁻⁴ OA 0005
4.8×10⁻³ OA 0006
8.6×10⁻⁵ OA 0007
1.2×10⁻⁵ OA 0008
8.8×10⁻⁵ OA 0009
1.2×10⁻⁵ OA 0010

Figure 3.25 SPC01/LOCA operator action tree; initiator: SPC01/LOCA.

4.0 THERMAL-HYDRAULIC EVALUATION

L. B. LaMonica, Science Applications, Inc.

4.1 Introduction

The role of thermal-hydraulic analysis in the probabilistic evaluation of PTS risks is to couple event tree analysis to the fracture mechanics (FM) analysis. Each end-state on an event tree defines a series of actions and/or failures that determines the response of the plant to the initiating event. The thermal-hydraulic response--measured as downcomer pressure, liquid temperature, and heat transfer coefficient--becomes the boundary condition used in the fracture mechanics calculation of crack propagation probability.

A thoroughly rigorous evaluation of PTS risk would require precise calculation of downcomer pressure, temperature, and heat transfer coefficient over the entire downcomer region for every event tree end-state. Since there are over 9.6 million end-states on the Oconee-1 event trees and detailed thermal-hydraulic simulation costs are great, a completely rigorous approach is not practical.

A first simplification of the scope of the analysis may be achieved by the use of a lower limit (10^{-6}/ yr) on event frequency to screen out a very large number of events which individually would not measurably contribute to overall risk. This screening process is described in Chapter 3.0. The remaining sequences are still too numerous for individual detailed evaluation. Furthermore, the development of pressure, temperature, and heat transfer conditions over the entier downcomer region (i.e., azimuthal and axial variations in the boundary conditions) would necessitate the use of TRAC or a revised RELAP5 model including multiple downcomer regions. Therefore, a combination of detailed calculations and engineering estimates was employed to define "single point" pressure, temperature, and heat transfer coefficient boundary conditions for all of the sequences. These single point profiles assume a uniform temperature, pressure, and heat transfer coefficient throughout the downcomer region. Where a range of temperatures could be predicated by TRAC or by extrapolation, the downcomer sector with the lowest temperature profile was used for the FM analysis.

The detailed calculations were performed by Fletcher, et al., at the Idaho National Engineering Laboratory (INEL) using the RELAP5 code[4.1] and by Ireland, et al., at Los Alamos National Laboratory (LANL) using the TRAC-PF1 code.[4.2] The models developed at INEL and LANL included the primary and secondary portions of the nuclear steam supply system, main and emergency feedwater trains, and certain control functions of the ICS. At the time of their creation, these Oconee-1 plant models were the largest and most complex models ever applied to the RELAP5 and TRAC-PF1 codes. The characteristics of these models are summarized in Section 4.2.

Oak Ridge National Laboratory identified 14 transient sequences for detailed evaluation. Tables 4.1 and 4.2 contain a brief description of each sequence simulated using RELAP5 and TRAC respectively. For purposes of comparison, the Oconee-3 turbine trip (revised), main steamline break, and pressurizer surge line small-break events were calculated using both RELAP5 and TRAC. More detailed descriptions of the sequences are provided in Section 4.3. The transient sequences were selected to include 1) major

Table 4.1 INEL Oconee-1 PTS overcooling transient calculations

Transient	Description
1. Oconee-3 turbine trip	Simulation of actual plant transient of March 14, 1980
2. Main steam line break	34-in. steam line break; all systems operate as designed; steam generators isolated at 10 min; unaffected steam generator refilled at 15 min; restart 1 RCP per loop 10 min after attainment of subcooling margin
3. Steam generator overfeed	Failure of feedwater pump trip on high steam generator level; all other systems operate as designed
4. Hot leg small break (PORV)	Pressurizer relief valve sticks open; ICS fails to run back main feedwater; primary coolant pumps remain on
5. Revised main steam line break	34-in. steam line break; all systems operate as designed; steam generators isolated at 10 min; unaffected steam generator refilled at 15 min; restart 1 RCP per loop immediately upon attainment of subcooling margin
6. Maximum sustainable steam generator overfeed	Reduced main feedwater pump speed to avoid pressure trips; feedwater pump trip on low steam line steam quality; all other systems function as designed
7. Turbine bypass failure at hot standby	Four turbine bypass valves fail open at hot standby; turbine bypass block valves closed at 10 min
8. Pressurizer surge line small (2.5-in.-diam.) break	All systems operate as designed
9. Reactor coolant pump suction small (2.5-in.-diam.) break	All systems operate as designed
10. Steam generator tube rupture	Affected steam generator isolated after 20 min

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Table 4.2 Los Alamos Oconee-1 PTS overcooling transient calculations

Transient	Description
1. Oconee-3 turbine trip	Simulation of actual plant transient of March 14, 1980
2. Main steam line break	34-in. steam line break; all systems operate as designed; steam generators isolated at 10 min; unaffected steam generator refilled at 15 min
3. Small-break LOCA (PORV stuck open)	Pressurizer relief valve sticks open; ICS fails to run back main feedwater; primary coolant pump trip
4. Turbine-bypass valve failure (one bank of two valves)	One bank of TBVs fails to reseat after opening a. SG level control fails b. SG level control does not fail c. RCP restart; HPI trottled
5. Turbine bypass valve failure (two bank of two valves)	Two banks of TBVs fail to reseat after opening. a. SG level control fails b. SG level control does not fail c. RCP restart; HPI trottled
6. Small-break LOCA (2-in. hot leg break)	2-in.-diam hole in pressurizer surge line; RCP trip; all systems operate as designed
7. Small-break LOCA (4-in. hot leg break)	4-in.-diam hole in pressurizer surge line; RCP trip; all systems operate as designed.
8. Rancho Seco-type transient	Initial loss of feedwater followed by runaway emergency feedwater to both steam generators

overcooling initiators such as steam line breaks, feedwater overfeeds, steam generator tube ruptures, and LOCA events and 2) assumed failures of controls and operator interventions that would otherwise limit system cooldown and repressurization. The intent of this selection process was to obtain accurate calculation of some bounding cooldown events. The results of the detailed calculations are summarized in Section 4.3.

The RELAP5 and TRAC codes do not predict the phenomena of temperature plumes and other related effects. These effects are important because they can yield downcomer temperatures significantly lower than the values predicted by RELAP5 or TRAC, which both assume perfect mixing. To address this question, the predicted cold leg and downcomer flows and temperatures were evaluated by T.G. Theophanous of Purdue[4.3] for potential of flow stratification effects. These results are summarized in Section 4.4.

Section 4.5 deals with the evaluation of pressure, temperature, and heat transfer coefficient profiles for PTS sequences retained following the event tree evaluation and screening process. The methods and criteria used to screen the set of remaining events for similarity grouping are addressed. The use of the detailed TRAC and RELAP5 data for direct extrapolation and in a simple energy balance cooldown model is also discussed. The results are presented for each major cooldown initiator, including turbine bypass valve failure at full power and at hot standby, PORV-sized LOCA events, main and emergency feedwater overfeed events, and steam generator tube rupture events. The summary of the detailed TRAC and RELAP5 calculations and the extrapolated sequences in Section 4.6 provides a key for connecting event tree results of Chapter 3.0 to the fracture mechanics results of Chapter 5.0.

4.2 Detailed Thermal-Hydraulic Models

4.2.1 RELAP5 Model Description

This section provides a summary of the RELAP5 model of the Oconee-1 PWR prepared at INEL. A detailed description may be found in Reference 4.1.

The RELAP5 Oconee-1 PWR steady state model is a detailed model of the Oconee-1 PWR power plant describing all the major flow paths for both primary and secondary systems, including the main feed train. Power-operated relief vales (PORV), safety valves, and the emergency core cooling system (ECCS) also are modeled. Secondary side features include turbine bypass and turbine stop valves, safety valves, and the emergency feedwater (EFW) system. The integrated control system (ICS) was another feature modeled. The model contained 220 volumes, 232 junctions, and 208 heat structures. The configurations of the full power and hot standby versions of the model were modified as necessary to match the specifications of the transient cases. A description of the primary system, secondary system, feed train, and the control system are presented in the following sections.

4.2.1.1 Primary System

The noding diagrams for the RCS primary are presented in Figures 4.1, 4.2, and 4.3.

The Oconee-1 PWR plant is a "2-by-4 configuration," i.e., two loops each containing one hot leg and two cold legs. The loops were designated as

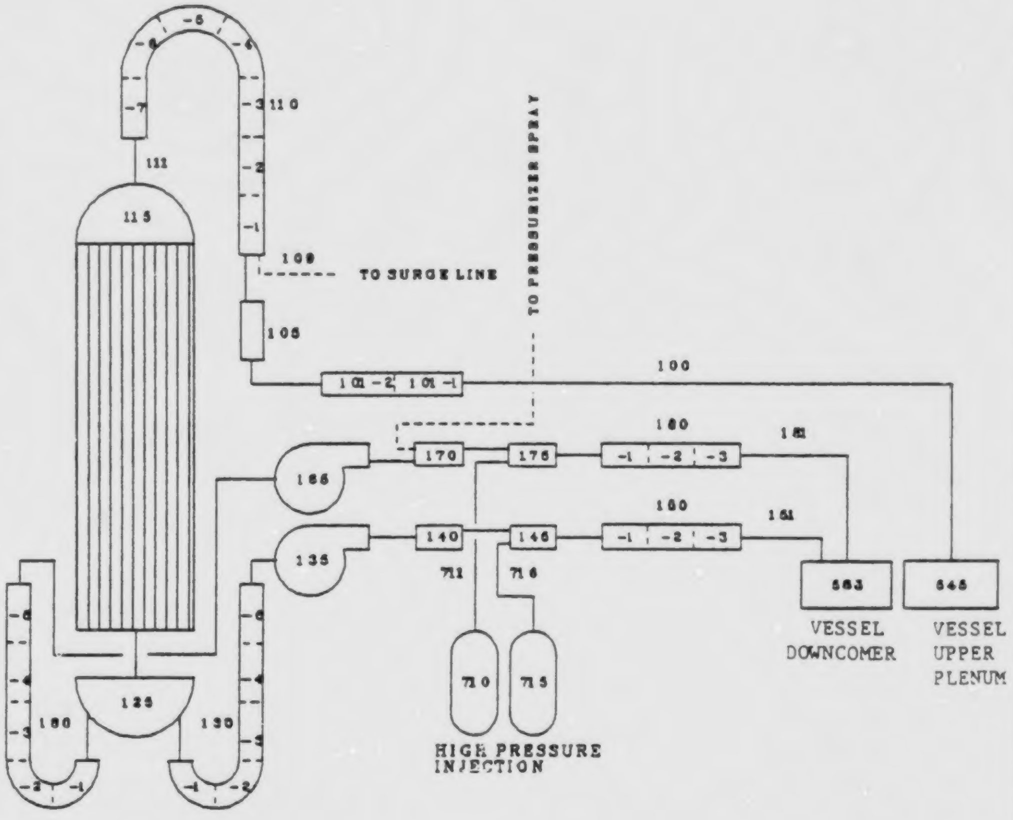
-5
-6
-4
-7
-3 110
111
-2
115
-1
109
TO SURGE LINE
TO PRESSURIZER SPRAY
105
101-2
101-1
100
180
181
170
175
-1
-2
-3
165
150
151
140
145
135
711
716
583
645
VESSEL DOWNCOMER
VESSEL UPPER PLENUM
-6
-4
-3
-2
-1
160
125
130
710
715
HIGH PRESSURE INJECTION

Figure 4.1 RELAP5 Oconee-1 model; primary Loop A.

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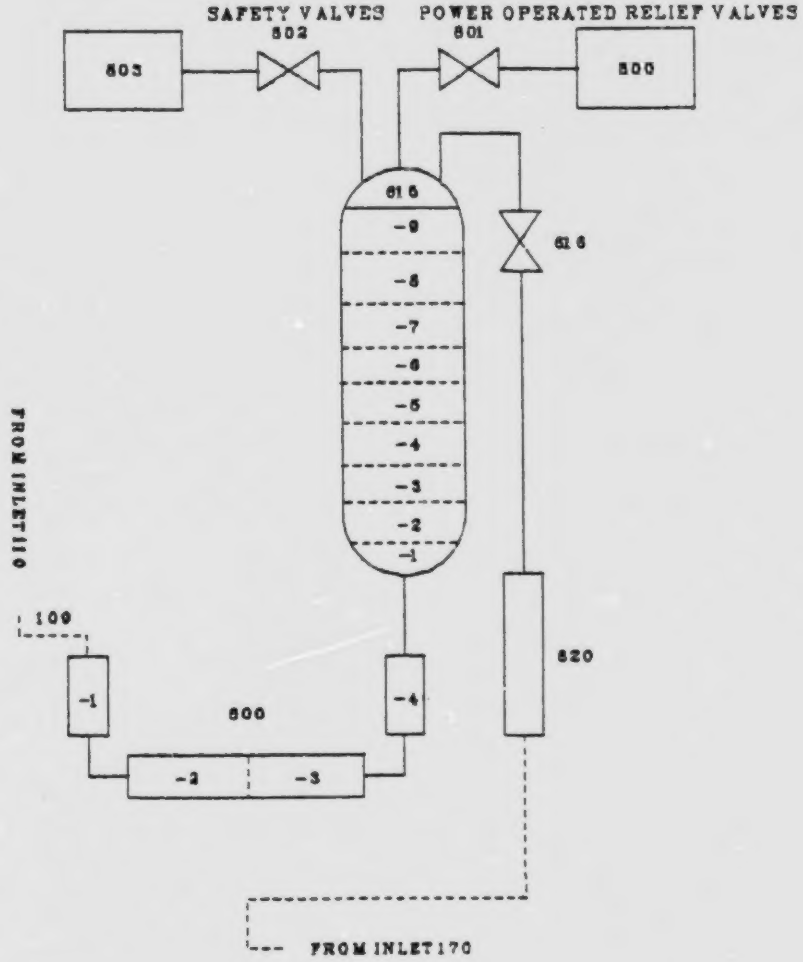

SAFETY VALVES
602
POWER OPERATED RELIEF VALVES
601
603
500
615
-9
-8
-7
-6
-5
-4
-3
-2
-1
616
FROM INLET 110
100
-1
600
-4
620
-2
-3
FROM INLET 170

Figure 4.2 RELAP5 Oconee-1 model; pressurizer system.

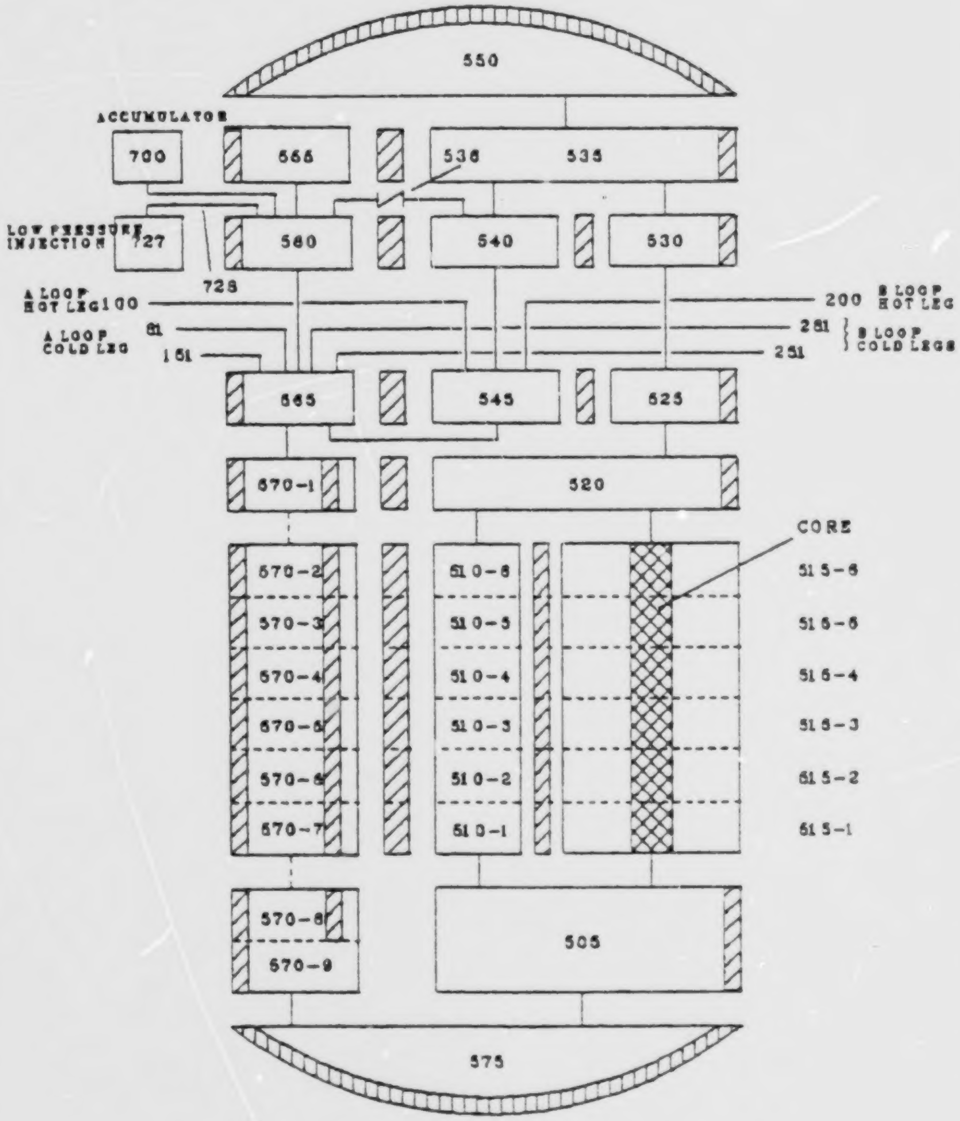
550
ACCUMULATOR
700
565
536
535
LOW PRESSURE INJECTION
727
728
560
540
530
A LOOP HOT LEG
100
200
B LOOP HOT LEG
A LOOP COLD LEG
81
151
281
251
B LOOP COLD LEGS
565
545
525
570-1
520
CORE
570-2
570-3
570-4
570-5
570-6
570-7
510-6
510-5
510-4
510-3
510-2
510-1
515-6
515-5
515-4
515-3
515-2
515-1
570-8
570-9
505
575

Figure 4.3 RELAP5 Oconee-1 model; vessel, core flood tank and LPI system.

loop A and loop B. Each loop contained one hot leg, one steam generator, two pump suction legs, two reactor coolant pumps, and two cold legs (refer to Figure 4.1). Loop B does not have the pressurizer surge and spray line connections, the only difference from loop A. The pressurizer surge line and pressurizer were attached to the hot leg in loop A (see Figure 4.2). Each cold leg contained a high-pressure injection (HPI) port.

The RELAP5 vessel model shown in Figure 4.3 consisted of several components describing the various vessel flow paths. The RELAP5 vessel included an inlet annulus, downcomer, lower plenum, core, core bypass, upper plenum, upper head, and a vent valve. The use of a single downcomer region prevented prediction of nonsymmetric temperature or flow effects due to the resultant mixing of the four cold leg streams.

The eight reactor vessel internal vent valves between the upper plenum and downcomer inlet annulus were modeled with a single RELAP5 servo valve. A servo valve uses the output of a RELAP5 control variable to determine the open area of the valve. The control variable was designed to calculate a valve flow area as a function of the differential pressure across the valve. Because the range of differential pressures over which the valve position changes is small, it was necessary to smooth the calculated differential pressure to prevent valve chattering. Without smoothing, the valve position responded instantaneously to the differential pressure across it. With smoothing, the change in valve position in a time step was limited, thus preventing excessively quick opening and closing of the valve and qualitatively simulating the response time of the flapper valves. Within the limitations of the thermal hydraulic models, the model of the vent valve used in the RELAP5 model is the best representation of the Oconee-1 reactor vessel vent valve presently available.

Also included in the primary system were the low-pressure injection (LPI) system, the core flood system, PORV, and safety valves. The LPI and core flood systems were connected to the vessel inlet annulus. The PORV and safety valves were connected to the top of the pressurizer. Table 4.2 summarizes the relationship between the physical components of the Oconee-1 LPI, core flood systems, PORV, and safety valves and the corresponding mathematical components of the RELAP5 model.

The RELAP5 nonequilibrium model was applied to all volumes in the primary system except for the accumulator where the accumulator model was applied. The wall friction model was applied in all primary volumes. Centrally located junctions for horizontal stratification were applied to all horizontal junctions. The choking and two-velocity models were applied at all primary junctions and an inertial solution was calculated at all junctions.

Heat structures were used to represent heat transfer and stored energy from fuel rods, steam generator tubes, loop piping, vessel wall, vessel internals, pressurizer wall, pressurizer surge line, and pressurizer heaters.

4.2.1.2 Secondary System

The RELAP5 model of the loop A steam generator secondary is shown schematically in Figure 4.4. Loop B has an identical noding structure.

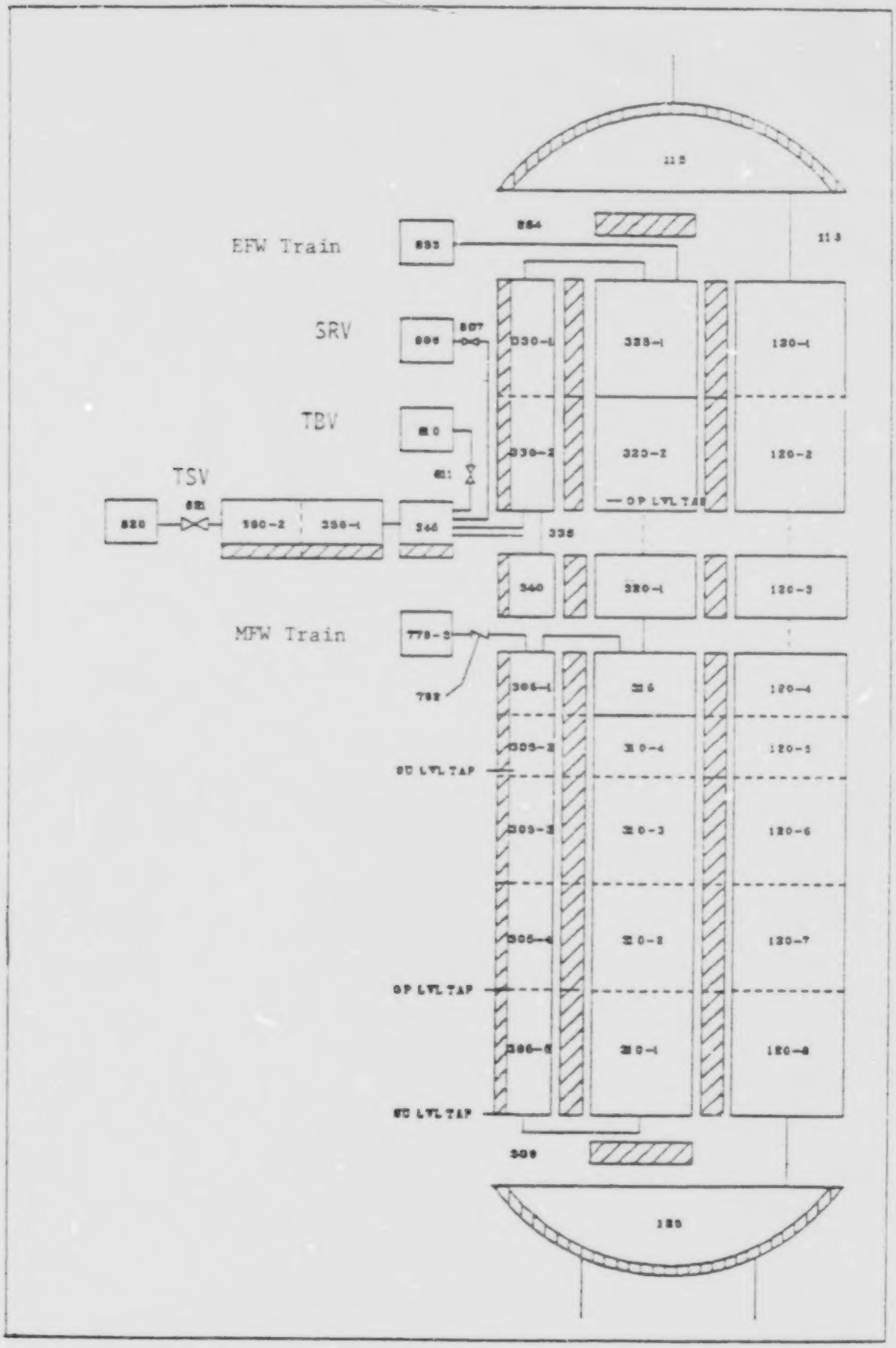
EFW Train
SRV
TBV
TSV
MFW Train
OP LVL TAP
SU LVL TAP
OP LVL TAP
SU LVL TAP
113
120-1
120-2
120-3
120-4
120-6
120-7
330-1
330-2
320-1
340
336
350-2
350-1
807
782

Figure 4.4 Loop A steam generator and steam line.

Originally the startup and operating levels were calculated based on the collapsed downcomer liquid level. Some early calculations were performed using this approach. For calculations presented in this report, the levels were based on calculated static pressures and fluid temperatures at the locations corresponding to pressure and temperature taps in the Oconee-1 steam generators. The calculated temperatures and differential pressures were smoothed in a manner consistent with the actual instrumentation system. The improved method of level calculation significantly affected the sequence of events in the INEL revised main steam line break transient, as discussed further in Section 4.3, and, in general, improved the behavior of emergency feedwater injection, which is controlled on level. The RELAP5 options used in the secondary system follow. The nonequilibrium option was applied to all volumes in the secondary system, a change from the equilibrium option used to perform the initial calculations with this model. Difficulties encountered with secondary behavior during the initial portion of the main steam line break calculation were overcome, allowing the nonequilibrium option to be used.

This change resulted in a more satisfactory behavior of the secondary system during periods of emergency feedwater injection. The wall friction model was applied to all secondary volumes. Centrally located junctions for horizontal stratification were applied to all horizontal junctions. The choking and two-velocity models were applied to all junctions, and an inertial solution was calculated at all junctions.

Heat structures were used to simulate the stored energy contained in the secondary shell wall, piping, and internals.

4.2.1.3 Feedwater System

The purpose of the feedwater system is to supply demineralized/deaerated subcooled water to the A and B steam generator secondary inlets. The water is heated and pressurized from 305 K (90°F) and 0.01 MPa (14.5 psia) to 511 K (460°F) and 6.55 MPa (950 psia). Figures 4.5 and 4.6 and Table 4.3 show the modeling arrangement of the feedwater trains.

The RELAP5 options used in describing the feedwater system include the nonequilibrium and wall friction models. Centrally located junctions were applied to all horizontal junctions, and choking and two-velocity models were also applied at all junctions.

Heat structures were used to represent the high- and low-pressure heaters and the piping wall metal masses. The feedwater heaters were modeled using the appropriate tube bundle surface areas. The energy contributed to the feedwater from the heater secondaries was modeled using the heat structure energy source option in RELAP5. The piping was modeled by first calculating the applicable metal volume, then adding the equivalent metal thickness to the appropriate heat structure. Heat structures representing several components were combined and added to selected components to reduce the computer storage requirements.

4.2.1.4 Control System

The Babcock & Wilcox integrated control system (ICS) is a comprehensive system that controls all power generation operations of the Oconee-1 plant. The models developed for the pressurized thermal shock program study of the Oconee-1 plant comprise only three portions of the overall

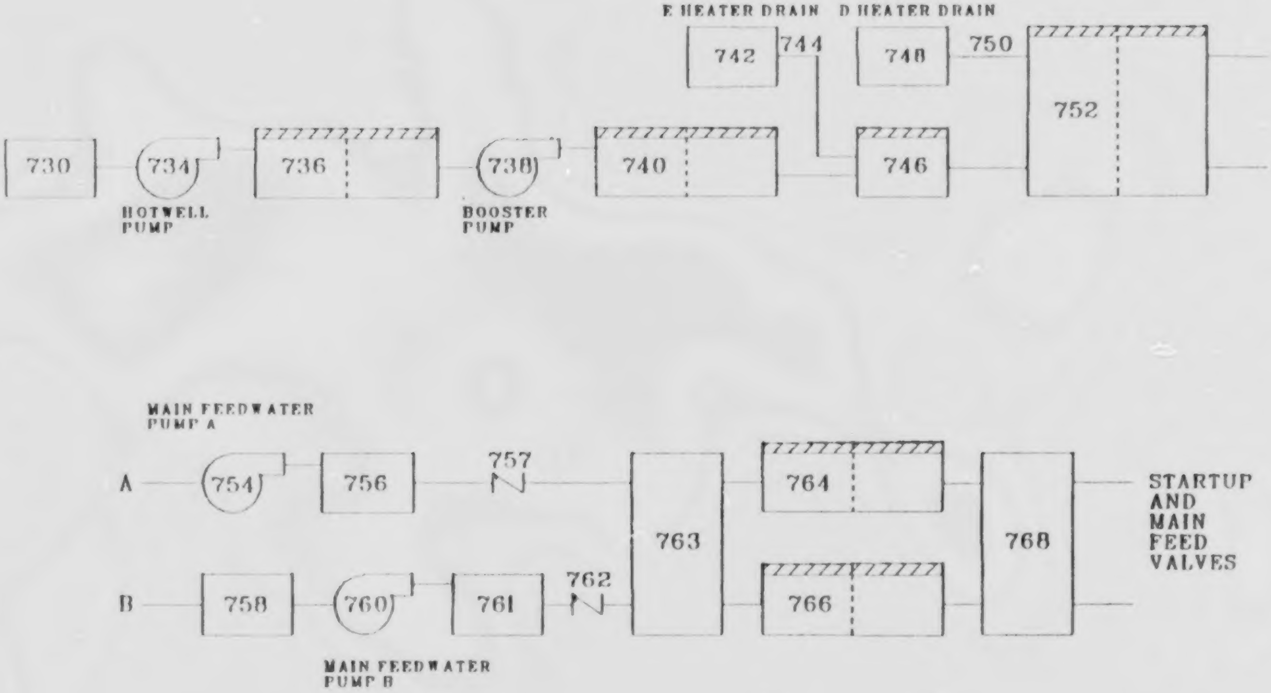
E HEATER DRAIN
D HEATER DRAIN
742
744
748
750
752
A
B
730
734
HOTWELL PUMP
736
738
BOOSTER PUMP
740
746
MAIN FEEDWATER PUMP A
A
754
756
757
763
764
768
STARTUP AND MAIN FEED VALVES
B
758
760
761
762
766
MAIN FEEDWATER PUMP B

Figure 4.5 RELAP5 Oconee-1 model; feedtrain from hotwell to startup and main feed valves.

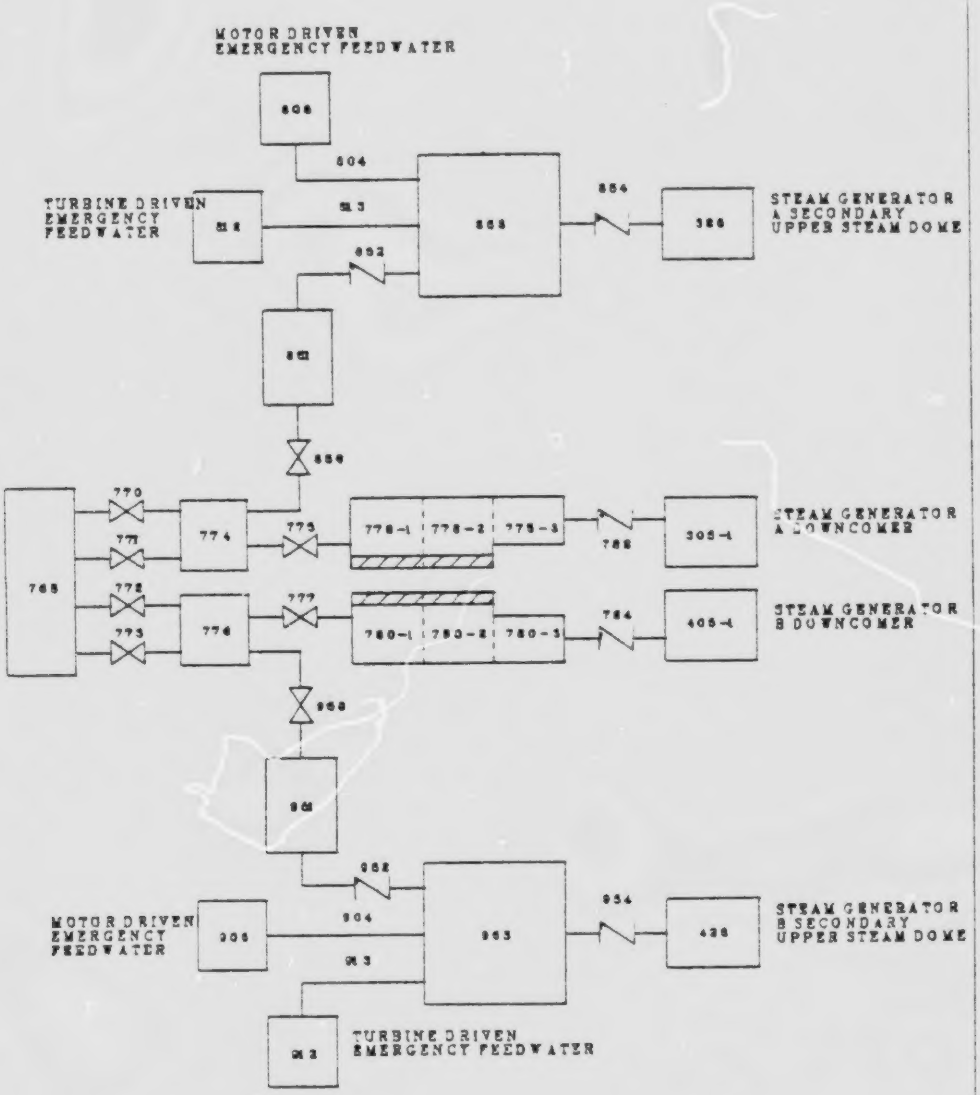
MOTOR DRIVEN
EMERGENCY FEEDWATER
806
804
TURBINE DRIVEN
EMERGENCY
FEEDWATER
812
813
853
854
386
STEAM GENERATOR
A SECONDARY
UPPER STEAM DOME
852
851
856
765
770
771
772
773
774
775
776
777
778-1
778-2
778-3
788
305-1
STEAM GENERATOR
A DOWNCOMER
780-1
780-2
780-3
784
405-1
STEAM GENERATOR
B DOWNCOMER
956
951
952
MOTOR DRIVEN
EMERGENCY
FEEDWATER
906
904
913
953
954
426
STEAM GENERATOR
B SECONDARY
UPPER STEAM DOME
912
TURBINE DRIVEN
EMERGENCY FEEDWATER

Figure 4.6 RELAP5 Oconee-1 model; feedtrain from startup and main feed valves to steam generator, crossover, and emergency feedwater system.

Table 4.3 Correspondence between physical components and the RELAP5 mathematical components in the Oconoee-1 feedtrain model

Physical Component	RELAP5 Component
Hotwell	730
Hotwell	734
Demineralizer and deaerator system	736
Booster pump	738
Low-pressure F and E heaters	740
E heater drain	742,744
Low-pressure D heaters	746
D heater drain	748, 750
Low-pressure C heaters	752
A main feedwater pump	754
A main feedwater pump discharge line	756, 757
B main feedwater pump inlet line	758
B main feedwater pump	760
Main feedwater pump discharge line	761, 762
Main feedwater pump header	763
A and B heater train number 1	764
A and B heater train number 2	766
High-pressure heater header	768
A and B startup and main feedwater control valves	770, 771, 772, 773
Steam Generator A control valve header	774, 775
Steam generator B control valve header	776, 777
Steam generator A feedwater header	778, 782
Steam generator B feedwater header	780, 784
Steam generator A feedwater crossover	850, 851, 852, 853, 854
Steam generator B feedwater crossover	950, 951, 952, 953, 954

control system: 1) turbine bypass control, 2) feedwater control, and 3) emergency feedwater control, the EFW control being separate from the ICS.

The model follows the description of the ICS given in Chapter 2.0, Section 2.2.3. The full complement of reactor protection trips and safety system actuation signals was included in the model.

4.2.2 TRAC Model Description

This section provides a summary of the TRAC model of the Oconee-1 PWR prepared at LANL. A detailed description may be found in Reference 4.2.

The Oconee-1 model developed for the PTS study represented the most comprehensive nuclear power plant modeling assembled to date for use with the TRAC code. The model contains a primary side, a secondary side, and a complex control system consisting of both trips and controllers.

4.2.2.1 Primary System

The noding diagram is presented in Figure 4.7, the primary side of the Oconee-1 model. It consists of the three-dimensional vessel, two hot legs, two once-through steam generators, four cold legs, and some parts of the emergency core cooling system (the low-pressure injection system was not modeled).

The volumes, elevations, and pipe lengths from the Oconee-1 plant are closely matched in the model. The wall areas and thicknesses of the primary piping also are modeled so that the thermal response of the piping is predicted. Heat transfer from the pipe walls to the environment also is allowed.

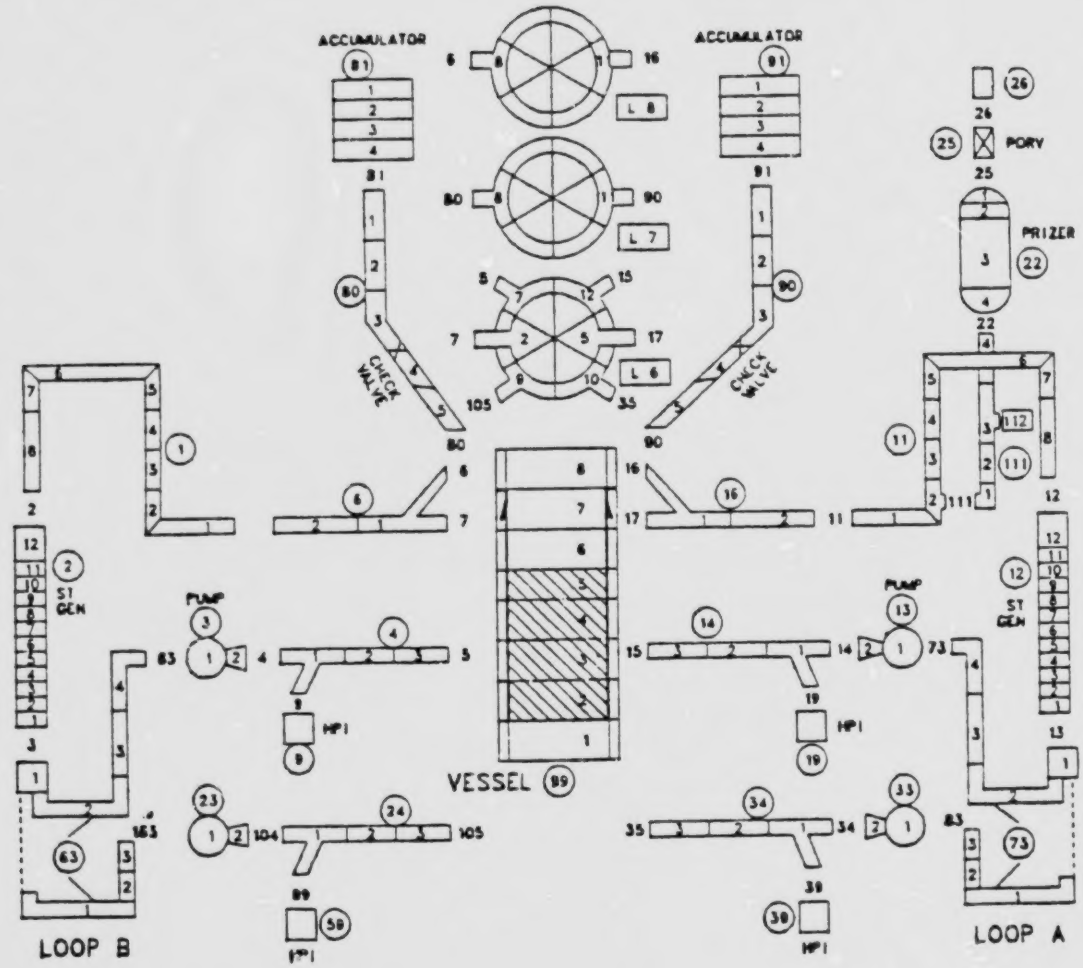
ACCUMULATOR
81
ACCUMULATOR
91
L 8
L 7
L 6
26
25
PORV
PRIZER
22
CHECK VALVE
CHECK VALVE
80
90
VESSEL 89
PUMP
3
PUMP
13
ST GEN
2
ST GEN
12
HPI
9
HPI
19
HPI
59
HPI
39
23
24
33
34
63
73
111
112
LOOP B
LOOP A

Figure 4.7 TRAC Primary-side model for Oconee-1.

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The three-dimensional vessel is made up of 96 cells arranged so that the vessel is divided into eight axial levels, two radial rings, and six azimuthal segments. The lower plenum is made up of one level, the core of four levels, the upper plenum of two levels, and the upper head region of one level. The two radial rings are divided so that the core region is bounded by the inner ring and the downcomer annulus is modeled in the outer ring.

The vessel metal structures and wall masses are modeled by assigning a representative thickness and area of a heat slab to each three-dimensional cell. Using these heat slabs, the stored energy of the vessel structure is approximated. The thermal conductivity for each heat slab is calculated by assigning five nodes across each slab thickness. These nodes are spaced so that they are closer together on the fluid side of the slab. There is no heat transfer through heat slabs between cells; that is, each cell's heat slab is isolated from any other cell's heat slab except through the fluid-dynamic coupling.

The six azimuthal divisions were chosen to allow for the angle and separation of the penetration of the hot and cold legs. Also, the accumulator injection ports were modeled close to their exact positions (both axially and azimuthally) in the vessel downcomer region.

With only one radial division for the inner portion of the vessel modeled, the circulation of hot water rising from the core into the upper head and descending out the hot legs is lost. Without finer noding of the vessel, the upper head is effectively isolated from the rest of the vessel in many circumstances. To alleviate this problem, a connection was assumed between the upper head and the hot legs so that about one-third of the normal steady state flow passes through the upper head. This upper-head tee connection influenced the movements of steam voids between the upper head and the candy canes of the hot legs. This behavior is most evident in the main steam line break sequence summarized in Section 4.3.

The vessel includes vent valves that are modeled to allow flow from the upper plenum directly to the downcomer. This flow path is only available when the upper plenum pressure is higher than the downcomer pressure. A composite vent valve made up of one-sixth the total vent-valve area is modeled in each azimuthal cell of the inner radius of axial level 7. The vent valves are fully open when the pressure drop between the upper plenum and downcomer exceeds 0.12 psi.

The loop A and loop B hot legs are modeled symmetrically except for the surge line connection to the loop A candy cane. The candy canes represent the highest elevation in the system. The surge line includes a "small-break" that is activated for the small hot leg break transients.

The pressurizer is modeled with a very small cell at the bottom and two small cells at the top. This is necessary to allow correct fluid conditions either to enter or leave the pressurizer.

Pressure relief for the primary system is provided by a single PORV at the top of the pressurizer. This valve supplies adequate relief for the cases in which secondary cooling is provided and the reactor has tripped.

A complete model was developed for B&W once-through steam generators. The primary side of the steam generator is made up of 12 cells, whereas the

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secondary side uses 26. Heat transfer from the primary occurs through the steam generator tubes to the secondary coolant. Cells 2 through 11 (Figure 4.7) represent a composite of the volume inside and wall area of all tubes, and it is in these cells that heat transfer from the coolant to the walls of the tubes occurs. Cells 1 and 12 model the upper and lower plenums. The wall area and thickness of the plenums are modeled so that the heat capacitance and heat transfer of the external wall can be calculated.

The secondary of the steam generator is divided into four components (Figure 4.8); two model the tube-bundle region, one models the downcomer, and the final one models the steam-outlet annulus. The tube bundle region is made up of 10 cells (SG components 12-1 and 12-2 for loop A and 2-1 and 2-2 for loop B in Figure 4.8). These cells model the total volume and the tube wall area for the region between all the steam generator tubes, but model the heat transfer characteristics of a unit cell of tubes. The top 4 of these 10 cells comprise the superheated-steam region of the model during normal full-power operation. The connection at the top cell of the tube bundle region is for the emergency feedwater (EFW) flow and the realigned main feedwater flow. This connection closely models the correct plant geometry so that any flow from the "upper header" is correctly injected into the tube bundle.

The steam generator downcomer consists of seven cells (SG components 12-3 and 2-3 of Figure 4.8) plus one cell for the main feedwater injection and one cell for the aspirator port. During normal operation, the downcomer condenses enough superheated steam drawn through the aspirator port to heat

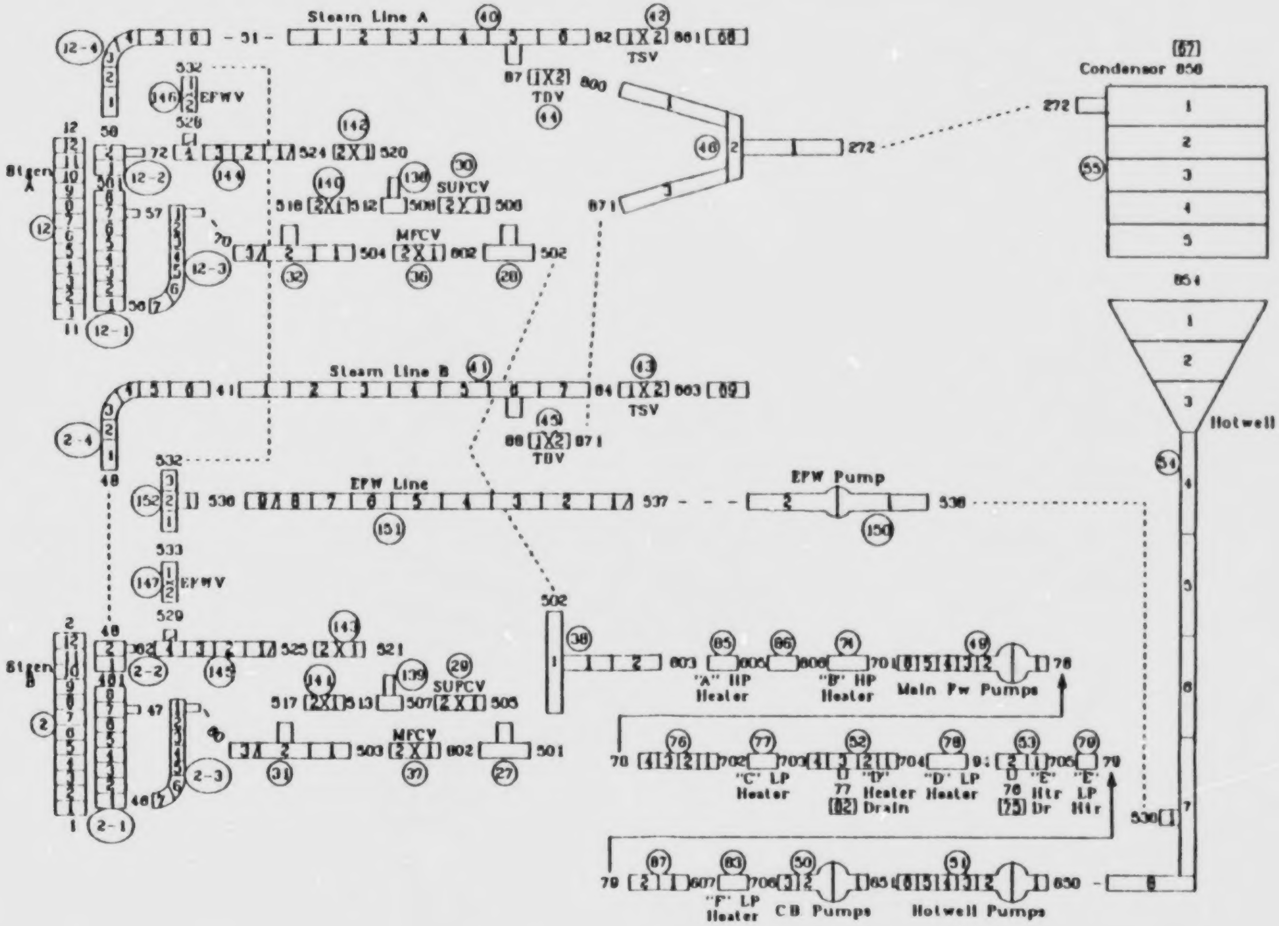
Steam Line A
Steam Line B
Stgen A
Stgen B
EFWV
TSV
TBV
SUPCV
MFCV
Condensor 858
Hotwell
EFW Line
EFW Pump
"A" HP Heater
"B" HP Heater
Main Fw Pumps
"C" LP Heater
"D" Heater Drain
"D" LP Heater
"E" Htr Dr
"E" LP Htr
"F" LP Heater
CB Pumps
Hotwell Pumps

Figure 4.8 TRAC Secondary-side model for Oconee-1.

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the main feedwater to saturation temperature prior to entry into the tube-bundle region. The final six steam generator secondary cells model the steam exit annulus. The outlet from this annulus is near the midpoint of the steam generator, close to the actual location in the plant.

The SG water levels were modeled using the best available methodolcgy, a collapsed liquid level calculation. This method is adequate for transients in which relatively slow changes in the secondary side occur, but it can generate wide discrepancies relative to the pressure tap calculation used in the RELAP5 model for rapid changes, such as in a MSLB transient.

All four cold legs of the plant were modeled. These legs each consist of a loop seal (lowest primary-loop point), a reactor coolant pump (RCP), and the HPI connection. The HPI nozzles are positioned closely to their correct height and distance from the vessel entrance. The RCPs are speed controlled during steady state operation to obtain the required mass flow, but their steady state speed is maintained fixed while they are running during a transient. During normal operation, the four cold legs of this model have symmetric flows.

Two major components of the ECCS included in this model were the HPI system and the accumulators (core flood tanks). If a transient were to be run that caused the primary pressure to fall below the low-pressure injection system (LPIS) set point, then the LPIS also would have to be added to the model.

The HPI system was modeled as four boundary conditions that can inject 283 K (50°F) water into the primary through the four side nozzles of the cold legs. These nozzles are located so they enter the main cold legs from the side at an elevation somewhat higher than the centerline of the cold/hot legs. The pressure-dependent flow rate of the two loop B ports is identical, as is also the case for loop A, but the loop A capacity is greater than that of loop B.

There is an accumulator tank for each loop that allows emergency coolant to flow directly into the downcomer region in axial level 7 of the vessel (Figure 4.7). The accumulator flow is controlled by check valves so that when the primary pressure falls below the accumulator-tank pressure, the check valves open. The initial accumulator pressure was ~4.2 MPa (~610 psia) with a coolant temperature of 305.4 K (90°F).

4.2.2.2 Secondary Side

All major components of the secondary side are modeled except for the turbine generator equipment and various valves that were not necessary for any transients of immediate interest. With the exception of the vessel, the secondary side required much more modeling detail than was necessary when only the primary system was modeled.

In this discussion, the feedwater train describes the secondary-side modeling from the condenser to the tee, where the feedwater is divided between the two steam generators. This section of the modeling takes the fluid discharge from the turbines and raises it to the temperature and pressure at which it is delivered to the steam generators.

The condenser is modeled as a large tank with a very large wall area. The thin walls have a high thermal conductivity and a constant temperature heat sink on the outside surface. This model fully condenses the incoming steam from either the turbine exhaust or the turbine bypass system.

The hotwell is an even larger tank used for the collection and storage of the condensate. The lowest system pressure and temperature occur in this component. Cell 1 of the hotwell actually represents the volume and coolant inventory of the upper surge tank. It is included to reduce the complexity of the model while still providing an estimate of the available hotwell inventory. The coolant supplied to the emergency feedwater system is taken from the hotwell/upper surge tank combination.

The hotwell pump (component 51) includes the demineralizer/aerator section of the feedwater train. The model accounts for the effects of this section by including additional frictional losses and heat addition to the coolant.

Each feedwater heater is modeled to achieve a feedwater temperature rise close to the design value for that heater. In addition, the model includes a time-dependent estimate of the feedwater-heater heat capacitance.

The two parallel MFW pumps of the actual plant are combined into a single pump for this model. This is a variable speed pump with the speed determined by the ICS.

The coolant flow from the feedwater train splits to provide flow to the steam generator-control valve section of each major loop. The loop A and loop B flow-control valves are identical. The ICS varies both MFW pump speed and feedwater control valve position to optimize control. Additional details of the model are available in Reference 4.2.

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The EFW system is modeled so that it takes water from the hotwell and delivers it to the EFWVs of both steam generators. The EFW pump is modeled as a composite of the turbine-driven and motor-driven pumps in the actual plant and therefore, has a large capacity. The EFW is delivered to a tee (component 152, Figure 4.8) that splits the flow between the two steam generators. If both EFWVs are open, the EFW flow is symmetrically split as long as the secondary pressure of the two steam generators is equal. If only one EFWV is open, the full flow available is delivered to that steam generator. The EFW flow stops if the hotwell inventory has been depleted.

The steam line for steam generator B is longer than that of steam generator A. Other than this difference, the steam lines and their valves are identical. A pressure boundary condition models the steam flow existing the secondary into the turbine inlet. During steady state operation, the steam mass flow is modeled to reenter the secondary as boundary condition inlets to the condenser and the two heaters drain. Following a turbine trip, the turbine stop valves (TSVs) close the steam lines. If any pressure relief from the closed steam lines is necessary, steam is released through the turbine bypass valves into the condenser. For a normal reactor/turbine trip transient, the steam relief from the turbine bypass valves is adequate so that modeling the main steam safety valves is not necessary.

4.2.2.3 Control Systems

The TRAC model included the integrated control system (ICS) functions discussed in Chapter 2.0, Section 2.2.3, namely, turbine bypass valve control, feedwater valve control, and main feedwater pump control. The

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full complement of reactor protection trips and safety system actuation signals was included in the models as described in Reference 4.2.

4.3 Results of Detailed Calculations

4.3.1 RELAP5 Calculations

The sequences simulated by the RELAP5 model are listed in Table 4.1. Detailed discussion of the transients and the results are available in Reference 4.1. Summary descriptions of the transients are provided here to provide a context for discussion of sequence extrapolation in Section 4.5. The results for the Oconee-3 feedwater transient are omitted in reference 4.1 and are likewise not included here. A summary of all transients discussed here and in Sections 4.3.2 and 4.5 is provided in Section 4.6 and Table 4.25.

4.3.1.1 INEL Original Main Steam Line Break

A description of the transient appears in Table 4.4. The sequence is identical to that for the revised main steam line break presented in Section 4.3.1.4 except for the reactor coolant (RCP) restart and high-pressure injection (HPI) throttling criteria. For this sequence, one RCP per loop is to be restarted 10 min after attainment of 28 K (50°F) subcooling in both hot legs, and HPI is not throttled. In the revised sequence, the RCP restart is immediate upon attaining 42 K (75°F) subcooling, and HPI is to be throttled to maintain 28 to 56 K (50° to 100°F) subcooling.

Results of the calculation and extrapolations to 2 h are shown in Figures 4.9 through 4.11. Figure 4.9 shows the reactor vessel downcomer pressure

Table 4.4 INEL original main steam line break transient scenario

---

1. Reactor trips, coincident with break of 0.86 m (34 in) steam line
2. Turbine trips; TSVs close
3. ICS functions as designed
4. Protection systems on hotwell, condensate booster; MFW pumps function as designed
5. HPI actuates at set point of 10.34 MPa (1500 psig)
6. Operator trips RC pumps 30 s after HPI actuation
7. EFW pumps start when low MFW pump discharge pressure is sensed[*]
8. MFW/EFW[*] system attempts to maintain 6.1 m (240 in.) steam generator level
9. Core flood tanks actuate at set point[*]
10. LPI actuates at set point[*]
11. Operator isolates feedwater to affected steam generator 10 min[**] into the transient
12. Operator restarts one RC pump in each loop 10 min[***] after attaining 28 K (50°F) subcooling in the hot legs
13. PORV opens at set point of 16.9 MPa (2450 psig)
14. SRVs open at set point of 17.2 MPa (2500 psig)[*]
15. PORV/SRVs reseat at their set points of 16.5 MPa (2400 psig)
16. Pressurizer goes water-solid
17. EFW surge tank capacity of 272544 liters (72,00 gal) is exhausted; the two motor-driven EFW pumps trip[*]
18. Turbine-driven EFW continues to draw from hotwell[*]

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[*] Event is phenomenologically dependent.

[**] Five times a typical time for operator action, based on simulator experience.

[***] Two times a typical time for operator action, based on simulator experence.

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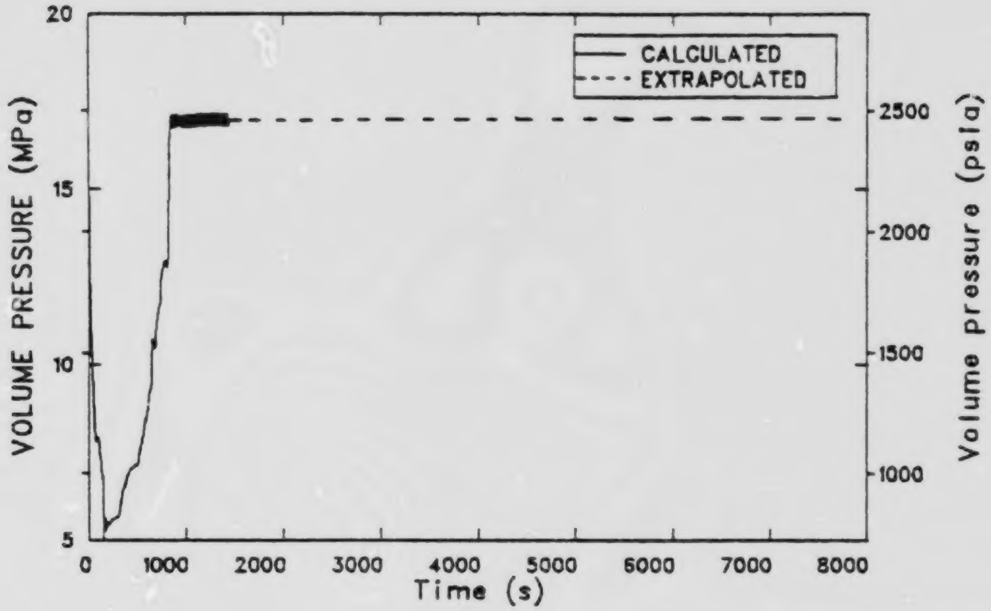

Figure 4.9 Main steam line break, RV downcomer pressure.

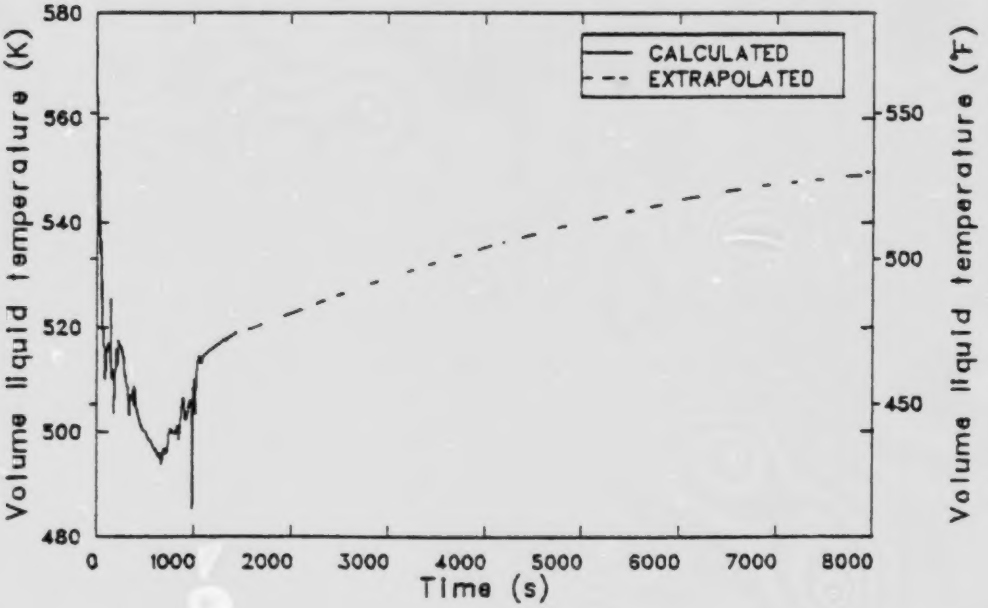

Figure 4.10 Main steam line break, RV downcomer fluid temperature.

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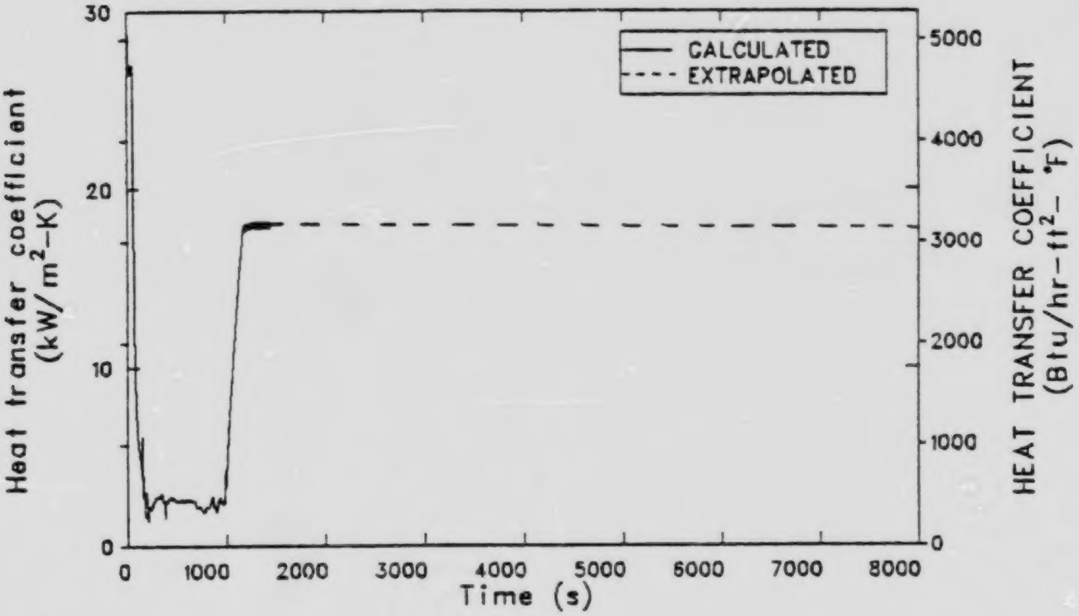

Figure 4.11 Main steam line break, RV downcomer inside surface heat transfer coefficient.

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response. The minimum pressure calculated was 5.02 MPa (728 psia) and the maximum was 16.99 MPa (2465 psia), the opening set point of the power-operated relief valve. Figure 4.10 shows the reactor vessel downcomer fluid temperature. The minimum temperature calculated was 482 K (407° F). The reactor vessel wall inside surface heat transfer coefficient is shown in Figure 4.11. The low values (for example, at 500 s) correspond to periods of natural circulation flow, and high values correspond to periods when reactor coolant pumps are operating.

4.3.1.2 Steam Generator Overfeed

A description of the transient is in Table 4.5. This sequence is transient number 53 (INEL3) in Table 4.25. The scenario differs from that of the maximum sustainable overfeed transient (presented in Section 4.3.1.5) in the availability of feed train protection functions. For this sequence, the protection functions, primarily low suction pressure trips on the feed train pumps, were available, and as a result the main feedwater pumps were tripped shortly after the transient was initiated. This sequence further differs from the sequence in Section 4.3.1.5 in that reactor coolant pump power is not tripped off by the operator.

Results of the calculation and extrapolations to 2 h are shown in Figures 4.12 through 4.14. Figure 4.12 shows the reactor vessel downcomer fluid pressure response; the minimum pressure calculated was 8.92 MPa (1293 psia). The maximum pressure calculated was 16.99 MPa (2465 psia), which corresponds to the power-operated relief valve opening set point pressure. The minimum calculated downcomer fluid temperature was 505 K (450°F), as shown in Figure 4.13. The reactor vessel wall inside surface heat transfer

Table 4.5 Steam generator overfeed transient scenario

1. Reactor trip, turbine trip, TSVs close
2. TBVs open and safety valves open and reseat at set points
3. Full MFW flow continues
4. HPI actuates at set point of 10.34 MPa (1500 psig), but operator does not trip RCPs
5. MFW pump trip on steam generator high level fails to occur
6. Steam generators fill
7. Core flood tanks actuate at set point[*]
8. LPI actuates at set point[*]
9. Protection systems on hotwell, condensate booster, and MRW/EFW pumps function as designed
10. Main steam lines fill with water
11. TBV closed at its set point on loss of condenser vacuum
12. EFW pumps start per low MFW pump discharge pressure set point
13. EFW flow controlled at steam generator level set point of 0.61 m (24 in)
14. PORV opens at set point of 16.9 MPa (2450 psig)
15. SRVs open at set point of 17.2 MPa (2500 psig)[*]
16. Pressurizer relief valves reseat at set point of 16.5 MPa (2400 psig)
17. Pressurizer goes water-solid

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[*] Event is phenomenologically dependent.

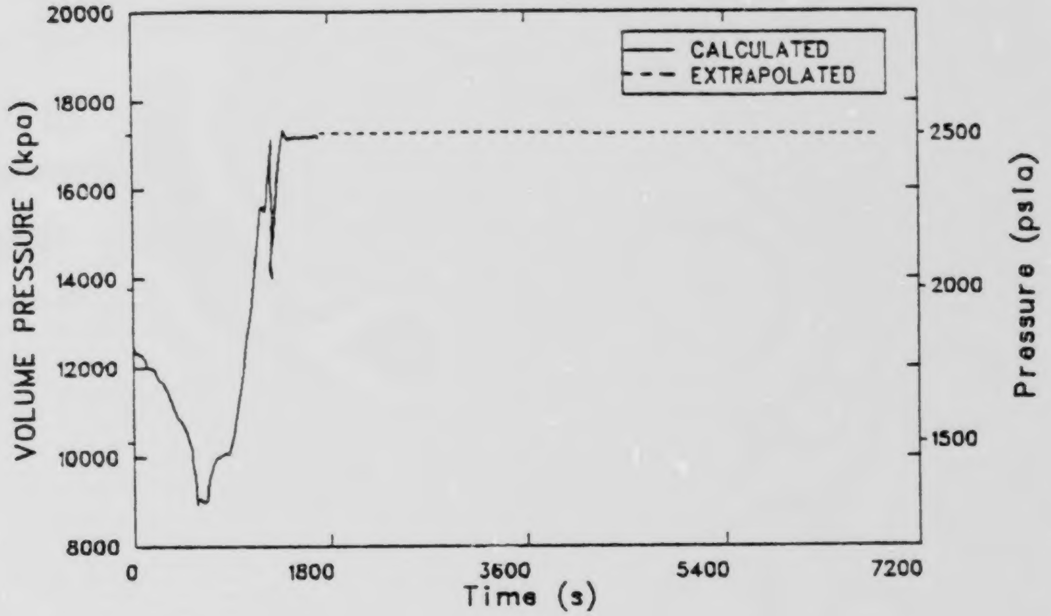

Figure 4.12 S.G. overfeed, RV downcomer pressure.

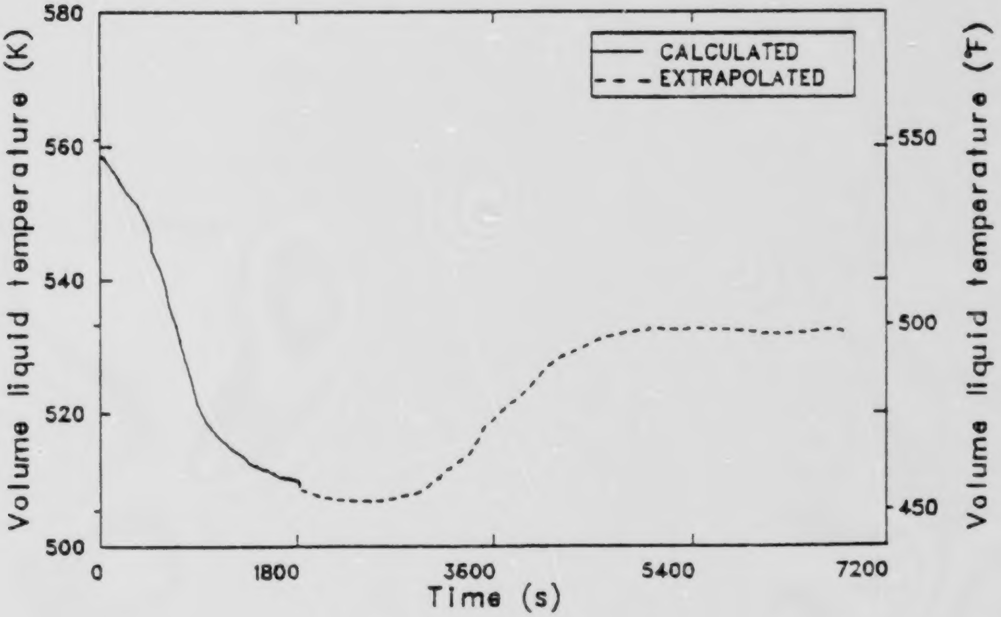

Figure 4.13 SG overfeed, RV downcomer fluid temperature.

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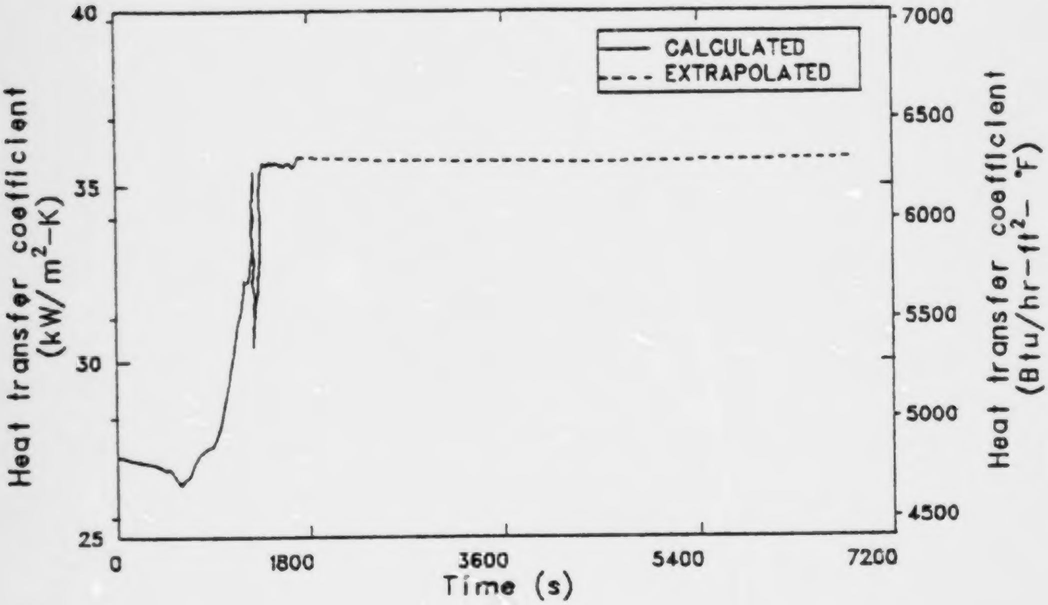

Figure 4.14 SG overfeed, RV downcomer inside surface heat transfer coefficient.

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coefficient, shown in Figure 4.14, represents forced convection conditions throughout the calculation. The variations observed in the coefficient are primarily due to changing fluid densities and pressures as the transient proceeded.

4.3.1.3 Hot Leg Small-Break Transient

The sequence is initiated by a stuck-open power-operated relief valve. The operator is assumed not to trip power to the reactor coolant pumps. The transient is described in Table 4.6. This sequence is transient number 52 (INEL2) in Table 4.25.

Results of the calculation and an extrapolation to 2 h are shown in Figures 4.15 through 4.17. Figure 4.15 shows the reactor vessel downcomer fluid pressure response. The minimum calculated pressure was 8.68 MPa (1259 psia). As shown in Figure 4.8, the minimum downcomer fluid temperature was 545 K (521^{o}F) and occurred at 2 h the end of the extrapolation. The extrapolated primary pressure at 2 h is 11.38 MPa (1650 psia). The heat transfer coefficient on the inside surface of the reactor vessel downcomer wall is extrapolated to drift downward as fluid properties change, as shown in Figure 4.17. The coefficient represents forced-convection conditions throughout the calculation.

4.3.1.4 Revised Main Steam Line Break

The revised main steam line break has the same specification as the LANL main steam line break base case (Section 4.3.2.1). This sequence is transient 51 (INEL1) in Table 4.25. The revised case differs from the original INEL case in the reactor coolant pump (RCP) restart and the high-

Table 4.6 Hot leg small-break transient scenario

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1. Small-break loss-of-coolant accident[*]
2. Reactor trips; turbine trips; TSVs close
3. HPI actuates at set point of 10.34 MPa (1500 psi)
4. TBVs/SRVs in secondary function as designed
5. ICS fails to run back MFW
6. MFW pumps trip on high steam generator level
7. EFW system functions as designed
8. Core flood tanks dump; LPI system actuates[**]

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[*] Size: 2.1368×10^{-3} m^2 (0.023 ft^2); location: pressurizer relief valve

[**] Event is contingent on size of break

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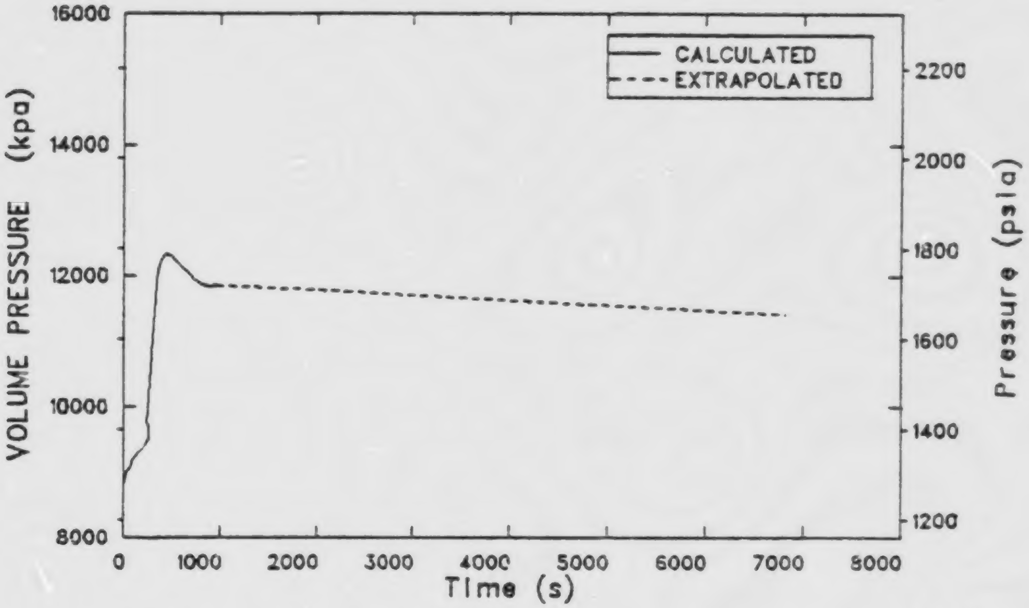

Figure 4.15 Hot leg small break, RV downcomer pressure.

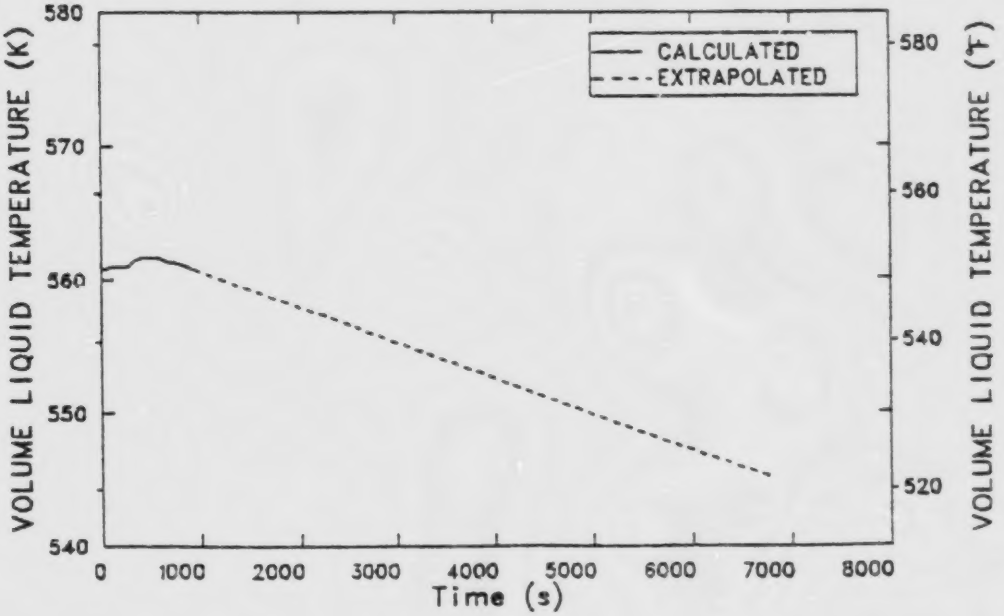

Figure 4.16 Hot leg small break RV downcomer fluid temperature.

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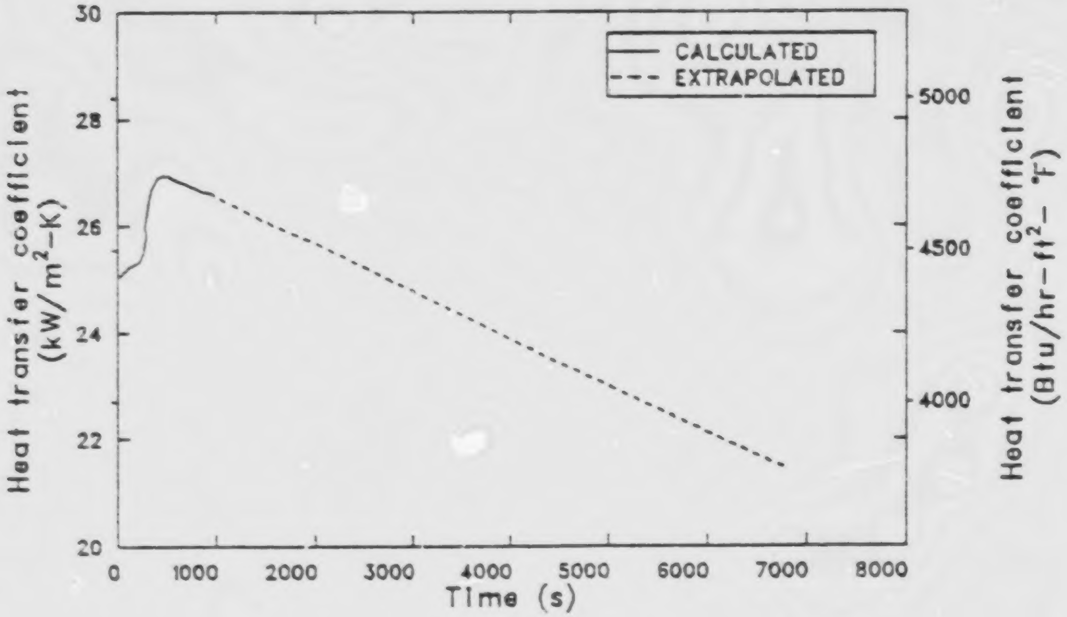

Figure 4.17 Hot leg small break, RV downcomer inside surface heat transfer coefficient.

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pressure injection (HPI) throttling criteria. In this case, the restart of one RCP per loop is immediate upon attaining 42°C (75°F) subcooling, and HPI is throttled to maintain subcooling of 42 ± 14°C (75 ± 25°F).

The sequence is initiated by a 200% double-ended rupture of a main steam line on one steam generator. All automatic plant functions are assumed to be operative. Operator actions are assumed to trip reactor coolant pump power 30 s after initiation of high-pressure injection, terminate all feedwater and turbine bypass on both steam generators at 10 min, and reactivate emergency feedwater and turbine bypass to the unaffected steam generator at 15 min.

Table 4.7 gives the sequence of events for the calculation. Figures 4.18 through 4.20 summarize the results of the calculation and extrapolation to 7200 s.

As shown in Figures 4.18 and 4.19, the calculated minimum reactor vessel downcomer mixed fluid temperature was 494 K (429°F) and the calculated maximum subsequent fluid pressure was 16.99 MPa (2465 psia). The downcomer heat transfer coefficient is given in Figure 4.20.

The calculation of downcomer fluid temperature was particularly sensitive to the requirement in the scenario to restart reactor coolant pumps upon attaining 42 K (75°F) subcooling in both hot legs. This requirement was met 300 s into the transient, and the restarting of the pumps caused the minimum downcomer temperature to be much warmer than if the pumps had not been restarted. Since the attainment of subcooling is sensitive to hot leg asymmetry and because a one-dimensional computer model such as RELAP5 will

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Table 4.7 Revised MSLB sequence of events

Event	Time from Start of Transient (s)
Reactor trip; turbine trip break opens	0
Main feedwater pumps tripped on high level in affected steam generator	0.3
Emergency feedwater tripped on, based on low main feedwater pump discharge pressure	4.4
HPI tripped on, based on low hot leg pressure	5.3
RC pumps tripped (30 s after HPI initiation) and main feedwater rerouted to emergency feedwater header	35.3
75°F subcooling attained in both hot legs. 2 RC pumps restarted, and HPI throttled to maintain 50-100°F subcooling	300
Unaffected steam generator level recovered to 240 in; main feedwater to this steam generator is terminated and emergency feedwater is throttled to maintain 240-in. level	320
Restarted RC pumps are up to full speed	360
Hotwell surge tank is empty; motor-driven emergency feedwater is terminated to both steam generators	513
Per scenario, all main and emergency feedwater and turbine bypass capability is terminated	600
Per scenario, turbine bypass and turbine-driven emergency feedwater is made available to the unaffected steam generator	900
Last time HPI is injected because subcooling is greater than 100°F after this time	1992
Pressurizer level reaches top	2354
PORV opening set point pressure reached	2432
Calculation terminated	2697

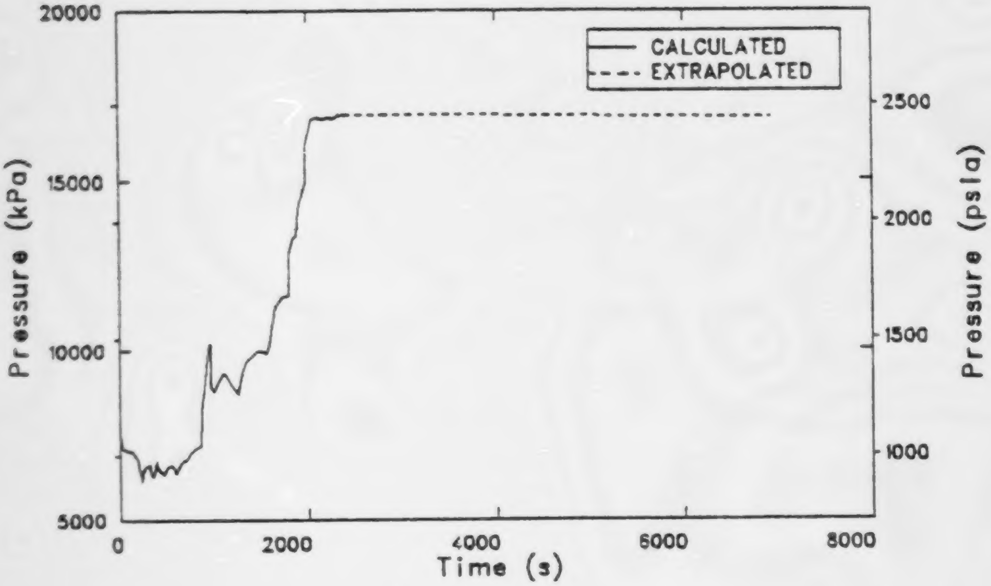

Figure 4.18 Revised MSLB, extrapolated pressure in RV downcomer.

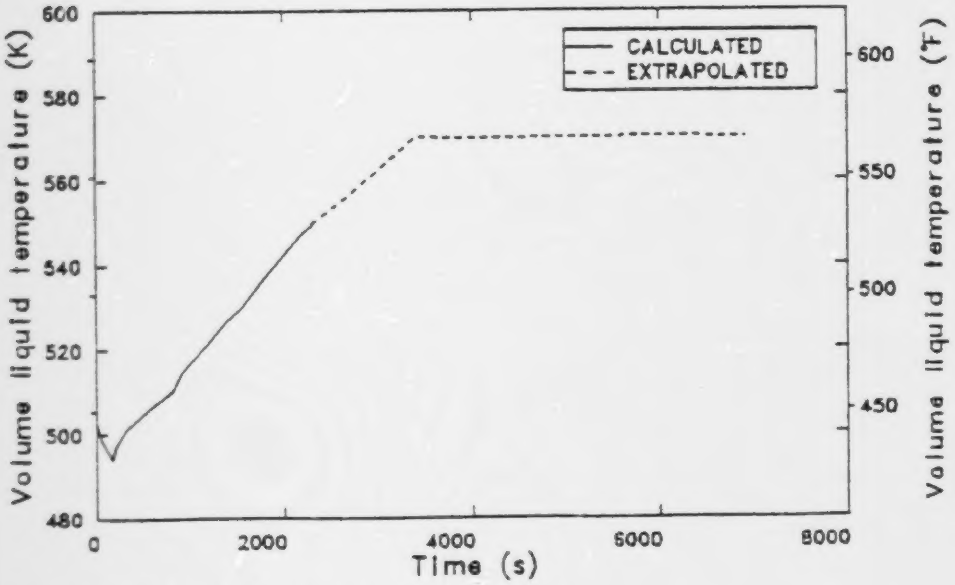

Figure 4.19 Revised MSLB, extrapolated fluid temperature in RV downcomer.

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not predict this asymmetry, the timing of the pump restart is uncertain. The effect of this and other specific uncertainties on minimum downcomer fluid temperature has been estimated to reduce the best estimate minimum downcomer fluid temperature to 462 K (372°F) with a lower uncertainty bound of 415 K (287°F).* See Reference 4.1 for further details.

4.3.1.5 Maximum Sustainable Overfeed Transient

The maximum sustainable overfeed transient employed a reduced MFW pump speed to circumvent the automatic feedwater system protection features (i.e., MFW pump trip on low suction pressure) that tripped the feedwater system early in the steam generator overfeed transient (Section 4.3.1.2). The maximum sustainable overfeed sequence is transient 56 (INEL6) in Table 4.25.

The transient was initiated from full-power steady state conditions (nominal temperature and pressure). The pressurizer heaters and spray operate as designed. The transient was initiated by a turbine trip that tripped the reactor and closed the turbine stop valves. Decay heat was assumed to be at the ANS standard rate. Instead of running back flow, the main feedwater system continued to supply the maximum sustainable flow to both steam generators without a trip from the component protection feature to the main feedwater pumps. It was assumed the main feedwater pump trip on steam generator high level failed to occur, allowing both steam generators to fill completely. The main feedwater pumps were eventually tripped on low steam quality in the main steam lines.

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*It should be noted that the adjustment of the downcomer temperature to reflect the uncertainty is inconsistent with the uncertainty analyses described later in this report.

Table 4.8 gives the sequence of events calculated for the transient. At the termination of the calculation, system behavior had been established such that extrapolation of the downcomer pressure, temperature, and vessel inside wall heat transfer coefficient to 7200 s was possible, as shown in Figures 4.21 through 4.23. The lowest downcomer liquid temperature in this time frame was 500 K (440°F). Subsequent to this minimum temperature, the maximum downcomer pressure was 16.99 MPa (2465 psia).

4.3.1.6 Turbine Bypass Valve Failure at Reactor Hot Standby

The turbine bypass valve failure at reactor hot standby is listed as transient 57 (LANL7) in Table 4.25. The RELAP5 full-power model was converted to hot standby conditions by the changes listed in Reference 4.1. As specified, the realignment of MFW to the emergency feedwater headers was locked out for this calculation.

The transient was initiated with the reactor at hot standby conditions. At zero time, all four turbine bypass valves (two on each steam line) are assumed to fail open, and all automatic plant systems are assumed to be operative. Operator actions are assumed for tripping reactor coolant pump power 30 s after high-pressure injection begins and closing turbine bypass block valves at 10 min.

Table 4.9 gives the sequence of events for the calculation. Figures 4.24 through 4.26 summarize the results of the calculation.

As shown in Figure 4.25, the minimum calculated reactor vessel downcomer fluid temperature was 387 K (237°F) near the end of the transient. The maximum calculated reactor vessel downcomer pressure was 16.99 MPa (2465

Table 4.8 Maximum sustainable overfeed transient sequence of events

Event	Time (s)
Turbine trip, reactor trip, turbine stop valves close, main feedwater pumps run back to 490 rad/s (4675 rpm); secondary pressure increased to turbine bypass and; safety relief valve set points. Feed train heater drains begin to close; pressurizer begins to drain	0.0
Heater drains closed	5.0
Steam generator B secondary pressure dropped below safety relief valve set point; valves close	25.0
Steam generator A secondary pressure dropped below safety relief valve set point; valves close	28.0
Pressurizer liquid level dropped below heater cutoff set point; heaters latched off	35.0
Steam generator B secondary completely liquid-full and steam line commenced to fill	240.0
Steam generator A secondary completely liquid-full and steam line commenced to fill	250.0
Primary pressure dropped below high-pressure injection set point and high-pressure injection flow was initiated	269.0
Vessel upper head began to void	275.0
Reactor coolant pumps trip; feed flow realigned to EFW header; main feed valves close	299.0
Steam generator B turbine bypass valves close	341.0
Steam generator A turbine bypass valves close	350.0
Void in steam line B dropped below 20%; main feedwater pumps trip off	417.0
Void in vessel upper head collapsed	434.0
Pressurizer began to refill	448.0
Primary system began to repressurize	524.0

Table 4.8 (continued)

Event	Time (s)
Steam generator A secondary pressure reached turbine bypass valve set point and valves crack open	930.0
Primary pressure reached power-operated relief valve opening set point and cycles around set point	1054.0
Pressurizer becomes liquid-solid	1079.0
Steam generator B secondary pressure reached turbine bypass valve set point and valves crack open	1424.0
Transient terminated; system pressure stabilized at power-operated relief valve opening set point; primary temperature was slowly increasing	1695.0

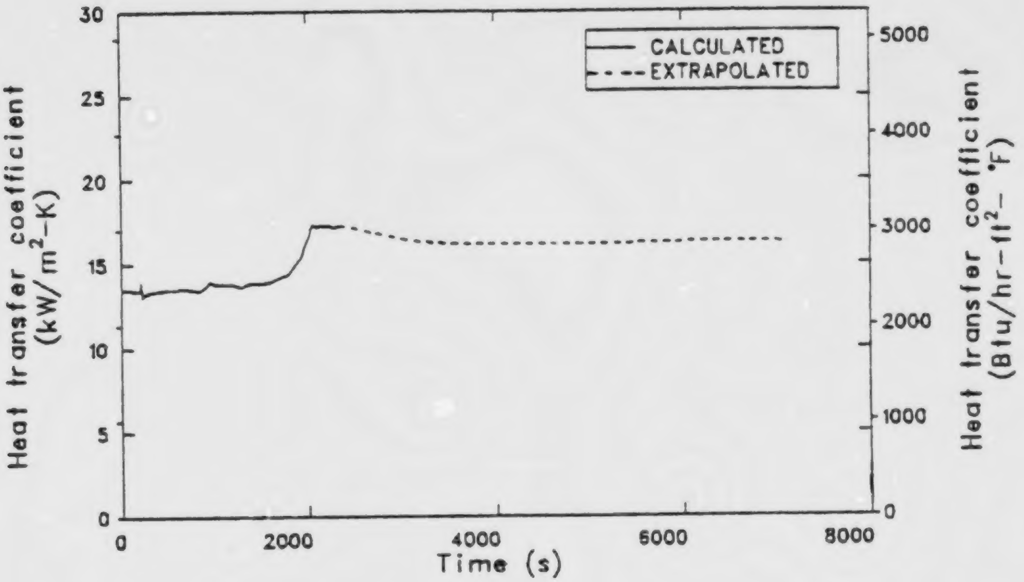

Figure 4.20 Revised MSLB, extrapolated heat transfer coefficient inside surface of reactor vessel downcomer wall.

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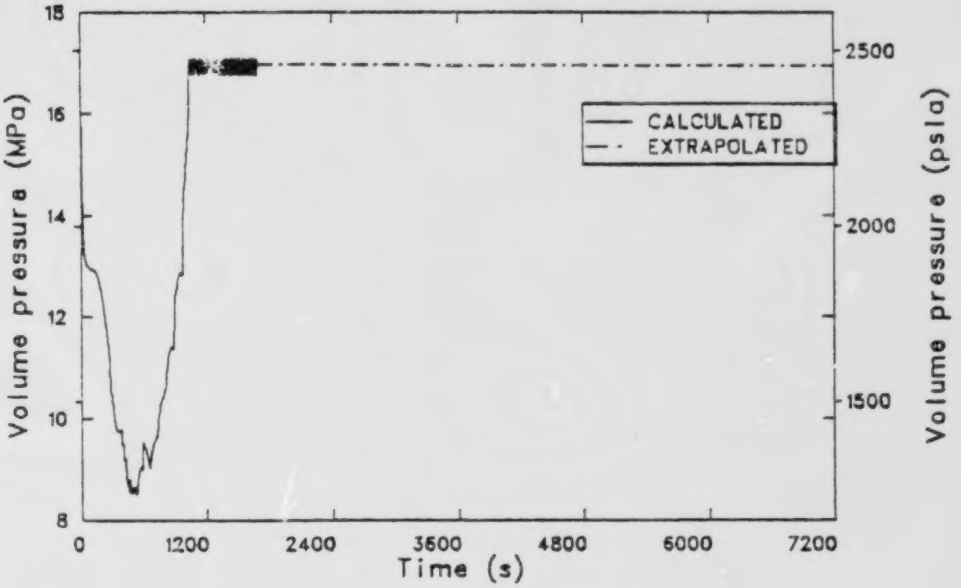

Figure 4.21 Maximum sustainable overfeed, reactor vessel downcomer fluid pressure.

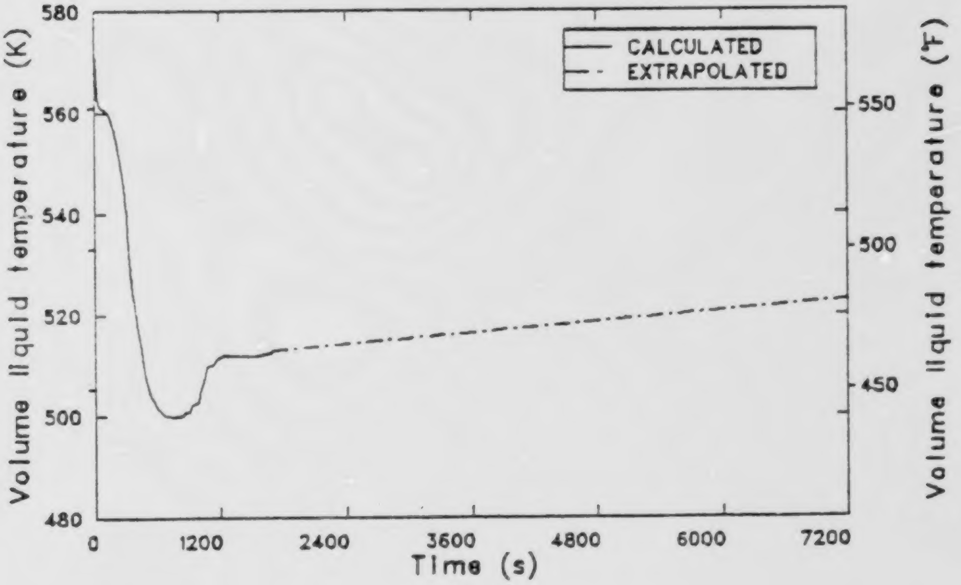

Figure 4.22 Maximum sustainable overfeed, reactor vessel downcomer fluid temperature.

Table 4.9. Turbine bypass valve failure at reactor hot standby transient sequence of events

Event	Time (s)
Turbine bypass valves fail open	0
MFW pump discharge pressure decreases to 765 psia	27.3
HPI initiated	125.1
RCPs tripped--steam generator low-level limit changed from 24 in. relative to SU level taps to 240 in. relative to operating level taps	155.1
EFW initiated in steam generator A and steam generator B because of change in low-level set point	
Condensate booster pump trips on low suction pressure	163.4
MFW pump trips on low suction pressure	168.5
MFW train isolated from steam generators after feedwater flow decreases to zero	228.6
Core flood tank injection initiated	383.5
Core flood tank injection ends	391.6
Turbine bypass valves closed	600.0
Pressurizer filled	900.0
PORV set point reached	949.6
Steam generator operating level at 240 in.	1010.2
Steam generator EFW sources tripped off for last time	1030.4
Steam generator A operating level at 240 in.	1070.5
Steam generator A EFW sources tripped off for last time	1074.6
Primary loop B cold leg suction temperature exceeds loop B hot leg temperature	1620.5
Primary loop A cold leg suction temperature exceeds loop A hot leg temperature	1892.2
Downcomer wall temperature reaches 265°F	4600.0
Transient terminated	7200.0

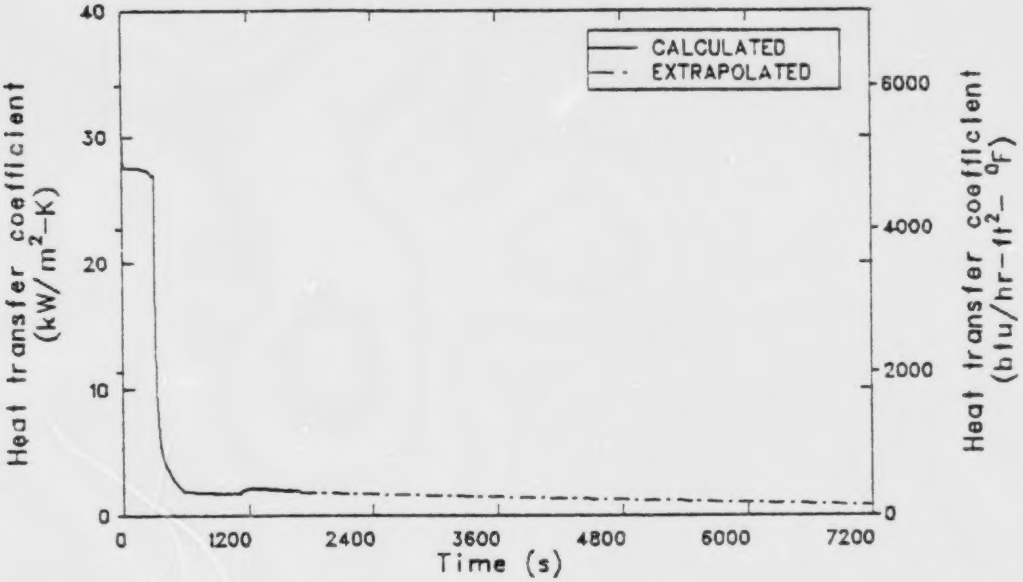

Figure 4.23 Maximum sustainable overfeed, reactor vessel downcomer inside surface heat transfer coefficient.

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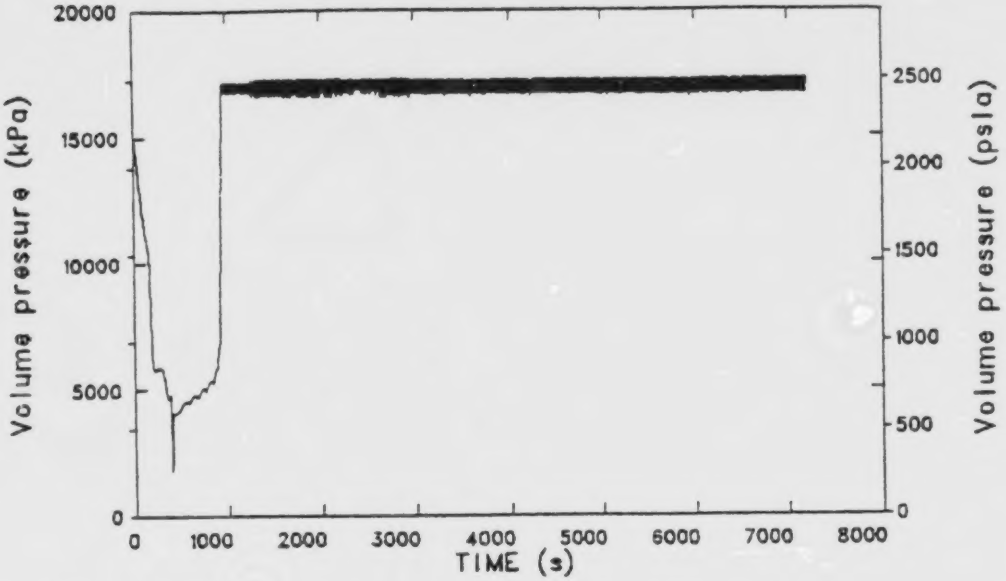

Figure 4.24 TBV failure at HSB, reactor vessel downcomer pressure.

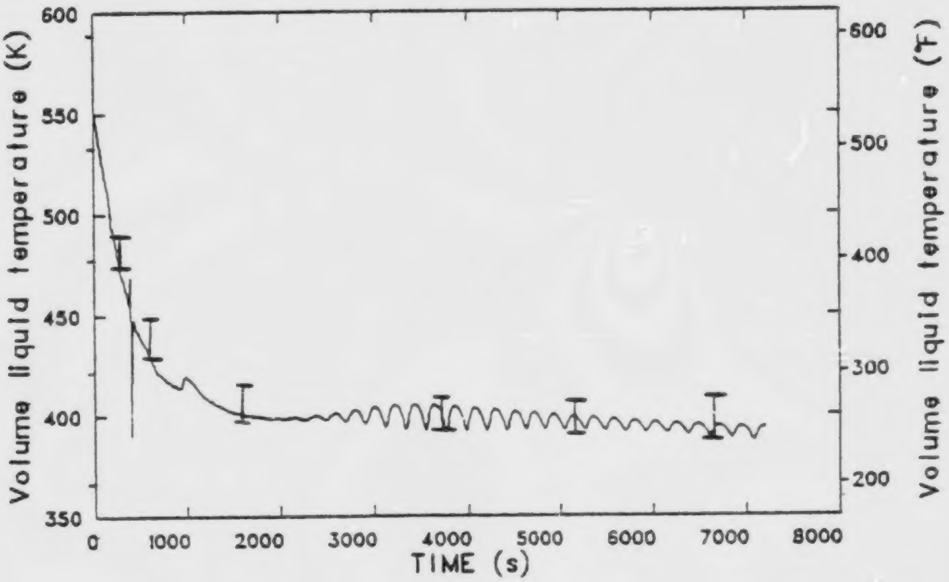

Figure 4.25 TBV failure at HSB, reactor vessel downcomer fluid temperature.

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psia), the opening set point pressure for the power-operated relief valve (see Figure 4.24).

The uncertainty in the calculated downcomer fluid temperature, expressed as error bars in Figure 4.25, was assessed. This uncertainty was due to startup valve operation, code-calculated condensation effects, and low level limit determination method. The uncertainty was found to be minor, and the conclusion is that the calculated temperature represents the lower temperature limit of the uncertainty band.

4.3.1.7 Pressurizer Surge Line Small-Break Transient

The pressurizer surge line small-break transient was calculated using both RELAP5 and TRAC (see Section 4.3.2.3). The INEL calculation is transient 55 (INEL5) in Table 4.25.

The transient was initiated by a 0.0508-m-(2-in.) diam break in the horizontal section of the pressurizer surge line with the reactor at full power conditions. Operator action is assumed to trip power to the reactor coolant pumps (RCPs) 30 s after initiation of high-pressure injection (HPI). This scenario is similar to that discussed in Section 4.3.1.3 except that the break is larger and has been relocated from the PORV to the surge line. Operator action to trip RCPs is assumed.

Table 4.10 gives the sequence of events for the calculation. Figures 4.27 through 4.29 summarize the results of the calculation and extrapolation to 7200 s. Error bars indicate the uncertainty limits for this calculation.

Table 4.10 Pressurizer surge line break transient sequence of events

Event	Time (s)
Open break in pressurizer surge line	0.0
Reactor scrams on P versus T relationship; turbine stop valves close; heater drains to feed train begin to close; MFW pumps begin to run back	45.2
TBV open on high steam generator pressure	47.0
SRV open on high steam generator pressure	50.0
Pressurizer heaters turn off on low-pressurizer liquid level	55.3
SRV close as steam generator pressure drops	69.0
MFW pumps trip on high pump discharge pressure	70.0
HPIS injection begins on low primary pressure	78.5
Upper head begins to void	86.0
RCP trip 30 s after HPI initiation; realign main feedwater to emergency feed headers; emergency feed begins on low steam generator level	108.5
TBV closes as steam generator pressure drops	117.0
Upper head volume completely drained of liquid; top volume of downcomer begins to void	300.0
Feed to steam generator B stopped as low liquid level limit is exceeded	500.0
Emergency feed to steam generator A stopped as low liquid level limit is exceeded	503.0
Vent valves open	554.0
Bubble forms in tubes of steam generator A; bubble starts to move to top of candy cane	768.0
Flow stagnates in loop A as bubble fills top of candy cane	815.0
Circular flow between cold legs of loop A	872.0

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Table 4.10 (continued)

Event	Time (s)
Bubble formed in tubes of steam generator B; bubble starts to move to top of candy cane	892.0
Flow stagnates in loop B as bubble fills top of candy cane	1020.0
Bubble in top volume of downcomer collapses	1383.0
Partial collapse of bubble in upper head; liquid drawn up is heated and vaporized by hot metal; primary pressure momentarily raises	1400.0
Bubble in upper head collapses, volume refills with liquid; large flows throughout primary system as liquid moves toward upper head	1488.0
Core flood tank begins injection on low primary pressure	2215.0
Circular flow between cold legs of loop B starts	2378.0
Circular flow between cold legs of loop B reverses direction	2443.0
Pressurizer begins to refill with liquid	5064.0
LPIS injection begins on low primary pressure	5124.0
Calculation terminated	6200.0

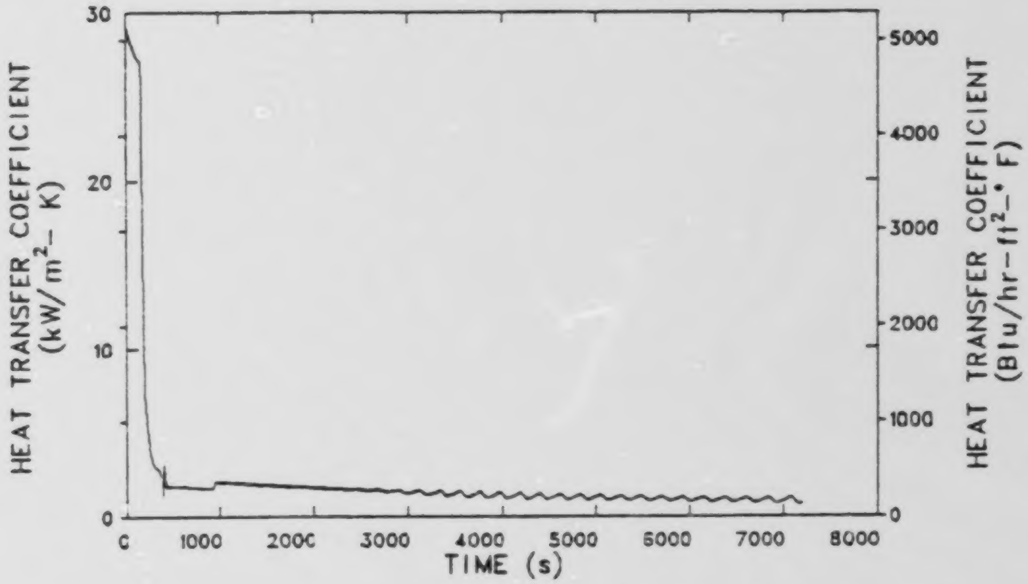

Figure 4.26 TBV failure at HSB, reactor vessel inside surface heat transfer coefficient.

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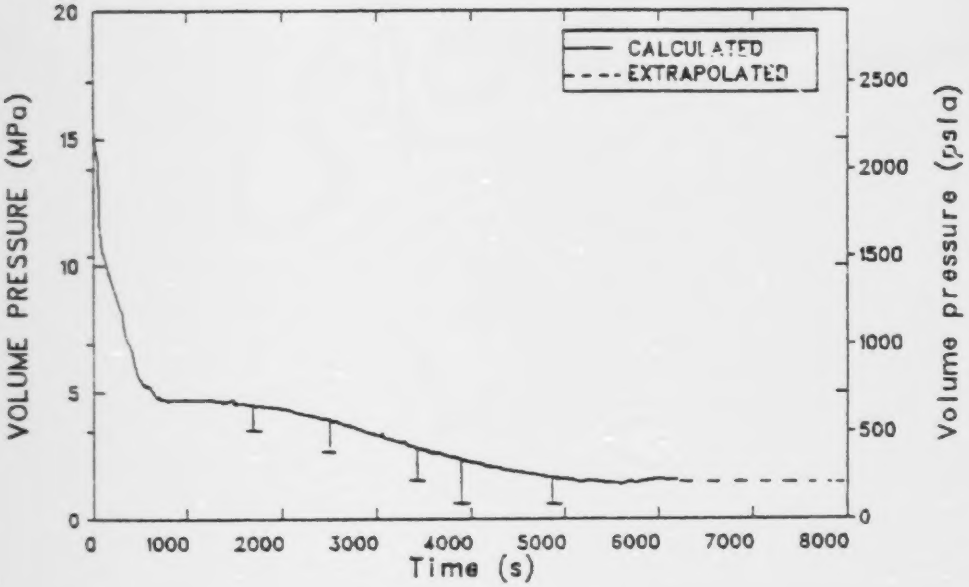

Figure 4.27 Pressurizer surge line break, extrapolated RV downcomer pressure.

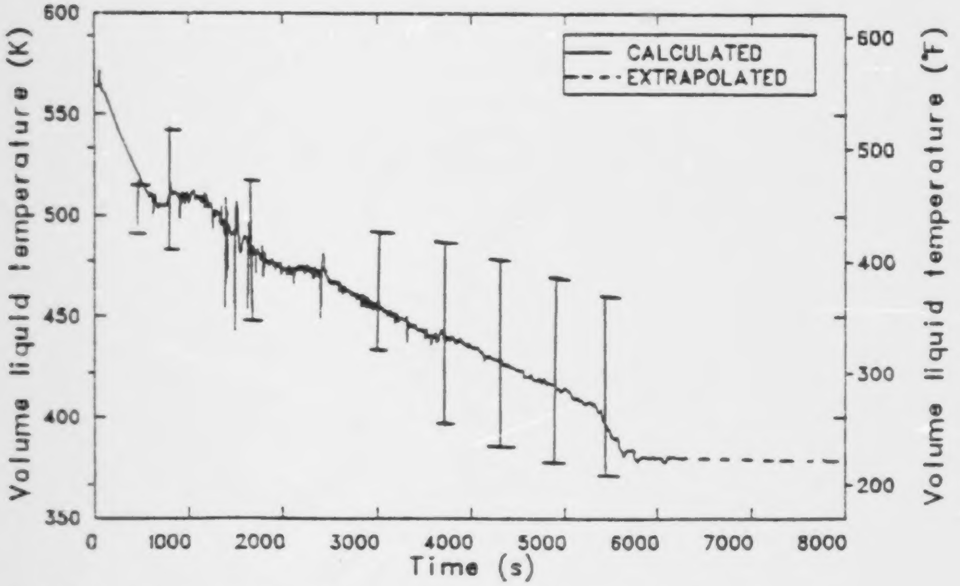

Figure 4.28 Pressurizer surge line break, extrapolated RV downcomer fluid temperature.

As shown in Figure 4.28, the minimum downcomer fluid temperature calculated was 380 K (225°F) and occurred at the end of the calculation, 6200 s. By extrapolating trends evident near the end of the calculation, the downcomer temperature would drop to 355 K (180°F) by 7200 s. The minimum cold leg temperature calculated (adjacent to the reactor vessel) was 333 K (140°F), which occurred at 1130 s when the primary system pressure was 6.1 MPa (740 psia). At the end of the calculation, the primary system pressure was at the LPI shutoff head, 1.48 MPa (214 psia). Since LPI injection was holding primary pressure at this point, it would be expected to stay very close to that pressure through 7200 s.

4.3.1.8 Reactor Coolant Pump Suction Small-Break Transient

The reactor coolant pump suction small-break sequence is transient 58 (INEL8) in Table 4.25.

The transient is initiated from full power steady state (nominal temperature and pressure). The pressurizer heaters and spray operate as designed. The transient is begun by a break downstream of a valve in the letdown line connected to the bottom of the A-1 pump suction leg.

The letdown line is 0.0635 m (2-1/2 in) in diameter. The break was assumed to occur downstream of a motor valve that could be isolated. Closure of this block valve was specified to occur only after core flood tank injection. The distance from the primary piping to the valve in the line was approximately 15.24 m (50 ft). The valve throat was assumed to have the same inside diameter as the pipe.

Following reactor scram, the decay heat is assumed to be the ANS standard. The turbine stop valves close, and it is assumed the integrated control system operates as designed. After the core flood tanks empty, the break is isolated and high-pressure injection flow is throttled when 28 K (50°F) subcooling is attained, but only if the primary system has repressurized to the power-operated relief valve (PORV) set point.

Table 4.11 gives the sequence of events for the calculation. Figures 4.30 through 4.32 summarize the calculation and extrapolation to 7200 s for downcomer temperature and pressure. Error bars in the figures identify the uncertainty limits. Calculated oscillations of cold leg flow contributed to the uncertainty and caused early termination of the calculation.

For the reactor coolant pump suction break sequence, the downcomer fluid temperature was 470 K (386°F) at the end of the calculation (4900 s). The transient was extrapolated to 7200 s (2 h), and at that time, the downcomer temperature was estimated to be 446 K (343°F) at a pressure of 5.1 MPa (740 psia).

The major uncertainty in the calculation is related to the existence of loop flow oscillations. Had these oscillations not been present in the calculation, the downcomer fluid temperature at 2 h would have been estimated to be 390 K (242°F).

The sequence requirement to isolate the break (by closing the motor block valve in the letdown) was not encountered because core flood tank injection did not occur. Had this isolation occurred, the subsequent maximum primary

Table 4.11 Pump suction break transient sequence of events

Event	Time (s)
Break in loop A-1 pump suction leg occurs	0.0
Reactor scrams on (P, T) relationship; turbine stop valves close; heater drains begin to close; turbine bypass and steam generator secondary safety valves operate as designed	48.60
Pressurizer heaters latched off due to low level in pressurizer	60.92
Main feedwater pumps trip on high discharge pressure	75.88
Primary pressure dropped below high-pressure injection set point and injection began	94.32
Upper head began to void	100.0
Reactor coolant pumps trip 30 s after high pressure injection initiation; main feedwater flow realigned to the EFW headers	124.0
Motor- and turbine-driven emergency feedwater systems initiated because of low level limit in the steam generator secondaries	124.36
Steam generator A and B turbine bypass valves close	135.0
Upper head completely voided	424.0
Steam generator B secondary pressure dropped below the booster pump head in the main feedline and main feedwater commenced to flow into A generator secondary, via the startup valves and crossover	500.0
Steam generator A secondary pressure dropped below the booster pump head in the main feedline and main feedwater commenced to flow into A generator secondary via the startup valves and crossover	545.0
Steam generator B emergency feedwater flow stopped because of liquid level above the low-level signal	573.71

Table 4.11 (continued)

Event	Time (s)
High level limit signal in steam generator B reached due to flow through startup valves	596.97
Loop natural circulation established	600.0
Steam generator A emergency feedwater flow stopped due to liquid level above the low-level signal	631.90
High-pressure injection volumetric flow rate exceeds the break volumetric flow rate and primary system depressurization essentially stopped	650.0
High level limit signal in steam generator A reached due to flow through startup valve	660.80
ICS ramped B startup valve closed, isolating steam generator B secondary	675.0
Vent valve flow commenced	680.0
Level in pressurizer began to recover; void in upper head began to collapse	700.0
ICS ramped A startup valve closed, isolating steam generator A secondary	731.0
Primary mass lost out the break regained by high-pressure injection mass	900.0
Upper head became liquid solid; primary system began to repressurize	1350.0
Natural circulation lost in loop B	2000.0
Primary pressure turned over; volumetric break flow exceeded volumetric HPI flow	2200.0
Flow in loop B reversed	2380.0
Flow reversal in loop B recovered	2800.0
Series of flow reversals in loop B mixed cold HPI water with warm primary liquid in the loop B cold legs	2800.0 to 4900.0

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Table 4.11 (continued)

Event	Time (s)
Transient terminated pattern established in transient to enable extrapolation of downcomer pressure and temperature to 7200 s; downcomer pressure and temperature at termination of transient were 6.0 MPa (870 psia) and 470 K (386°F), respectively	4900.0
Extrapolated downcomer pressure and fluid temperature are 5.1 MPa (740 psia) and 446 K (343°F) respectively	7200.0

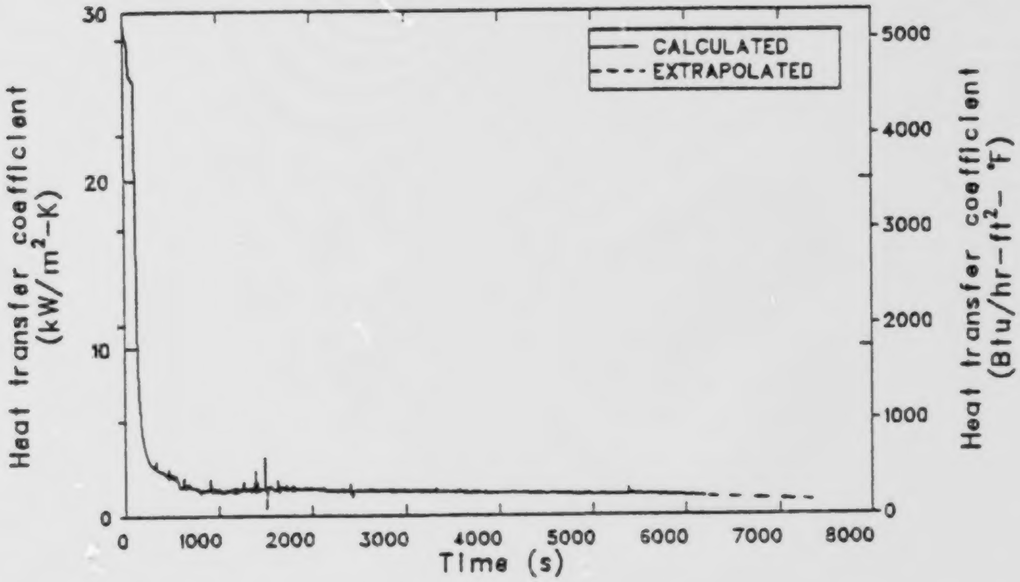

Figure 4.29 Pressurizer surge line break, extrapolated RV downcomer inside surface heat transfer coefficient.

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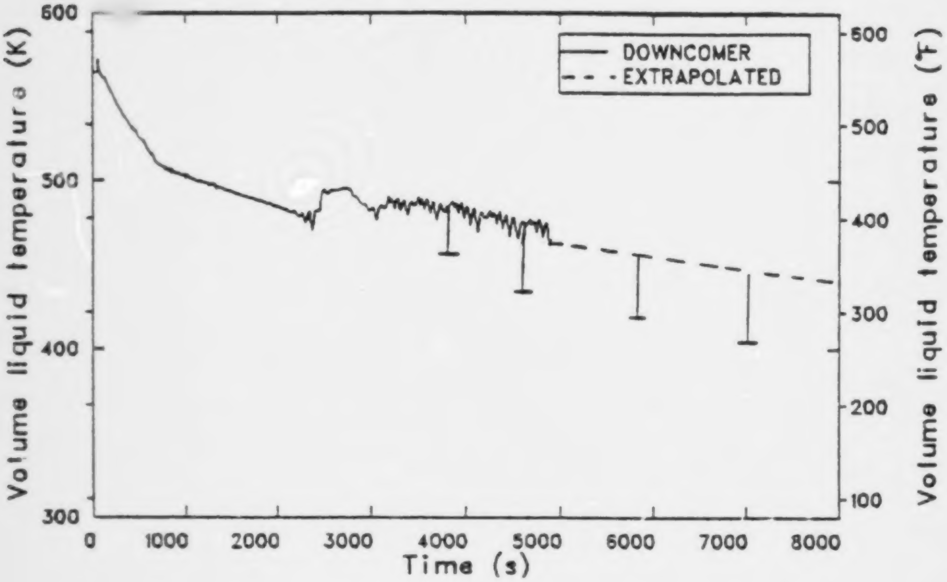

Figure 4.30 Pump suction break, reactor vessel downcomer fluid temperature.

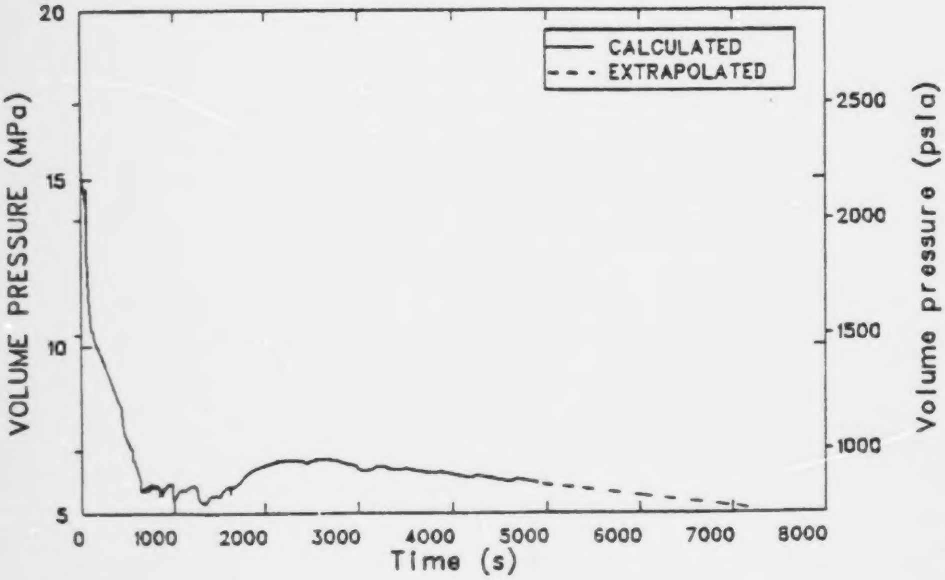

Figure 4.31 Pump suction break, reactor vessel downcomer fluid pressure.

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pressure would have been ~16.99 MPa (2465 psia), the opening pressure set point of the power-operated relief valve.

4.3.1.9 Steam Generator Tube Rupture Transient

The steam generator tube rupture sequence is transient 59 (INEL9) in Table 4.25. The transient is initiated from full power steady state conditions (nominal temperature and pressure). The pressurizer heaters and spray operate as designed. The transient begins with a rupture of a single steam generator tube (double-ended break) in the A loop steam generator. It is assumed the ICS operates as designed. The reactor scrams and the turbine stop valves close. Emergency feedwater starts on low-level limit signal and secondary safety relief and turbine bypass valves operate as designed.

Table 4.12 gives the sequence of events for the calculation. Figures 4.33 through 4.35 summarize the calculation and extrapolation to 7200 s. At the termination of the transient, the primary system pressure was oscillating around the PORV set point pressure of 17.0 MPa (2465 psia). The reactor vessel downcomer liquid temperature, which is of PTS concern, was at 517 K (470°F) and increasing slightly. As shown in Figures 4.33 and 4.34, the extrapolated values of pressure and temperature to 7200 s are 17.0 MPa (2465 psia) and 544 K (520°F) respectively. The lowest vessel downcomer liquid temperature was 505 K (450°F), which occurred at 1700 s.

4.3.2 TRAC Calculations

The sequences simulated by the TRAC model are listed in Table 4.2. Detailed discussion of the transients and the results are available in Reference 4.2. A summary of all transients discussed here and in Sections 4.3.1

Table 4.12 Steam generator tube rupture transient sequence of events

Event	Time (s)
Steam generator tube rupture in steam generator A occurs; ICS adjusts MFW valves to compensate for perturbation in steam generator A secondary system	0.0
Reactor scram (P/ T) relationship turbine trip, turbine stop valve closes, heater drains close over 5-s period; ICS runs back MFW pumps and MFW valves	319.2
Pressurizer heaters come on because of low primary pressure	321.6
Pressurizer level drops below 220 in. and makeup flow increases	325.4
Pressurizer levels drops below 127.6 in., cutting off the pressurizer heater power	336.3
Main feedwater pumps trip on high discharge pressure because of ICS running back the startup valves	349.0
Primary pressure increased because of reduction in primary to secondary heat transfer	650.0
Level in steam generator B secondary oscillates around low level set point, allowing periodic flow of emergency feedwater into secondary system; periodic flow of emergency feedwater enhances primary to secondary heat transfer, and primary pressure turns over	775.0
Primary pressure dropped below HPI set point, HPI flow initiated, makeup and letdown flow stopped	942.4
RCPs trip 30 s after HPI initiation; steam generator low-level set point increased and motor- and turbine-driven emergency feedwater system turned on; increased primary to secondary heat transfer enhances primary fluid cooldown	972.4
Liquid in the main feedline commenced to flow into steam generator B through the startup valve and crossover due to the secondary pressure dropping below the booster pump head	1280.0
Liquid in the main feedline commenced to flow into steam generator A through the startup valve and crossover because the secondary pressure dropped below the booster pump head	1300.0

Table 4.12 (continued)

Event	Time (s)
Liquid level in steam generator A rose above the level limit set point and EFW into the secondary was terminated	1396.6
Liquid level in steam generator B rose above the low level limit set point and EFW into the secondary was terminated	1415.5
Liquid level in steam generator A rose above the high-level limit set point and ICS closed the startup valve, isolating steam generator A	1426.0
Liquid level in steam generator B rose above the high-level limit set point and ICS closed the startup valve, isolating steam generator B	1437.6
Steam generators become heat sources; primary temperatures begin to increase	1600.0
Primary pressure reached PORV set point; the valves oscillate around the set point, maintaining primary pressure	2200.0
Pressurizer becomes liquid-solid	2450.0
Transient terminated; vessel downcomer liquid temperature at 517 K (471°F) and increasing; pressure at the PORV set point	2800.0

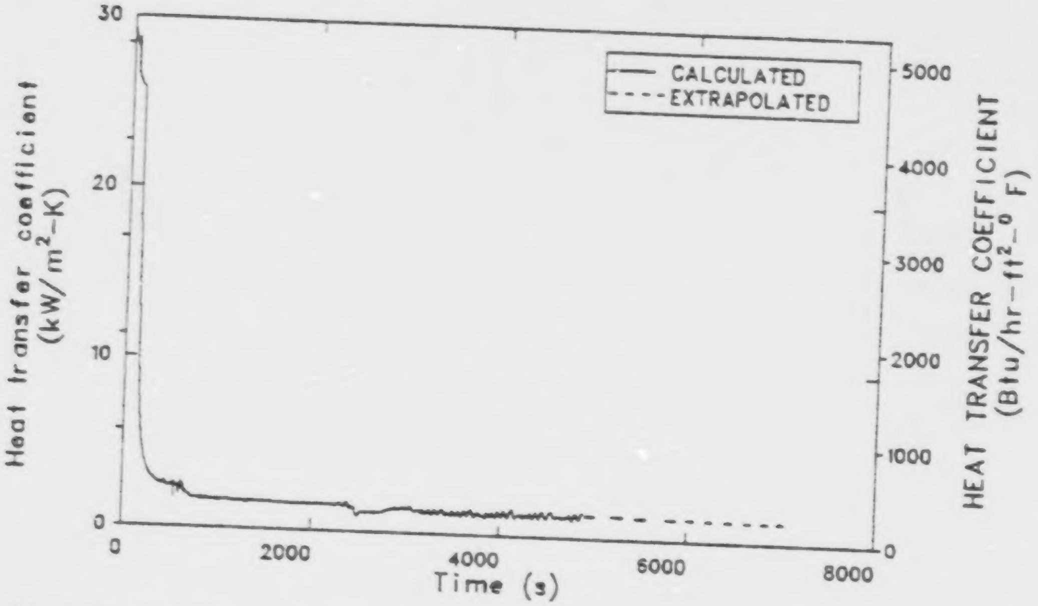

Figure 4.32 Pump suction break, reactor vessel downcomer inside surface heat transfer coefficient.

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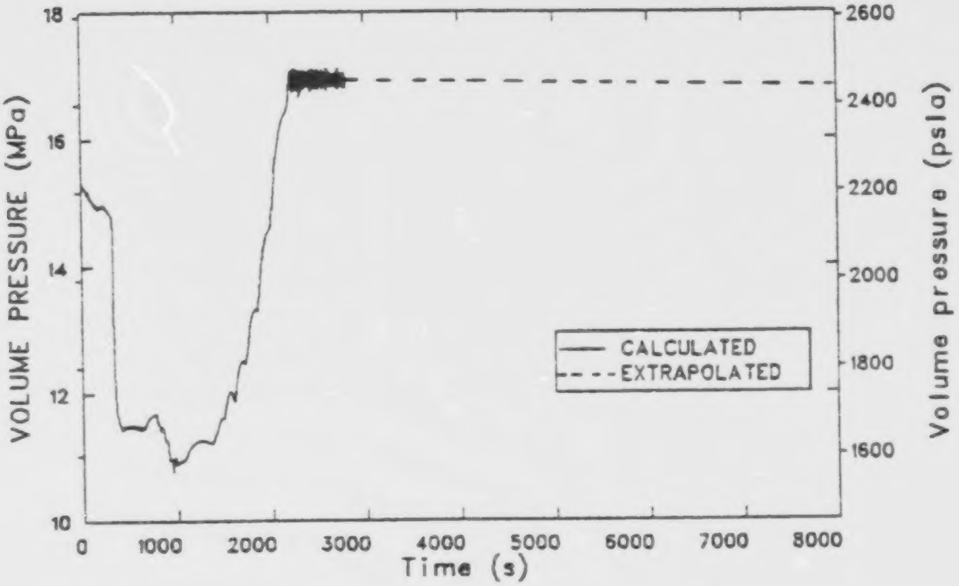

Figure 4.33 SG tube rupture, extrapolated RV downcomer fluid pressure.

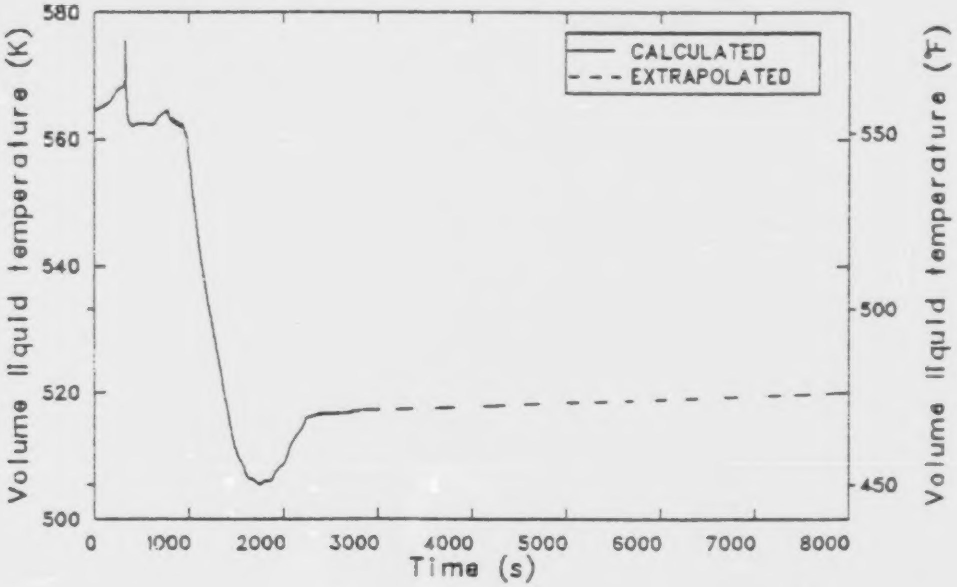

Figure 4.34 SG tube rupture, extrapolated RV downcomer fluid temperature.

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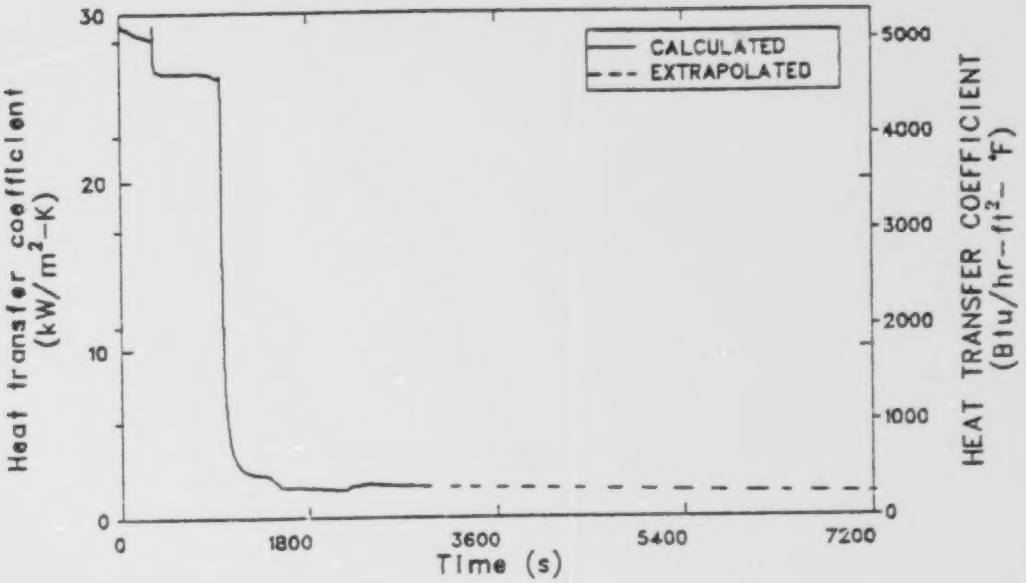

Figure 4.35 SG tube rupture, extrapolated RV downcomer inside surface heat transfer coefficient.

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and 4.5 is provided in Section 4.6 and Table 4.25. The Oconee-3 feedwater transient was a benchmark case of no direct importance to this study and is omitted here. Reference 4.2 contains a discussion of that calculation.

4.3.2.1 Main Steam Line Break

The LANL main steam line break (MSLB) base case is the TRAC model calculation (Section 4.3.1.4). Three parametric cases (cases 2, 3 and 4) reflecting different feedwater system responses were also performed. The MSLB base case and parametric cases 2 and 3 appear in Table 4.25 as transients 35 (LANL1), 37 (LANL3), and 38 (LANL4) respectively. Parametric case 4 was provided too late for inclusion in all phases of this study, but would significantly impact the extrapolation of steam line break sequences (see Appendix C, Section C.3.) The present discussion centers on the MSLB base case calculation.

The transient is initiated by a double ended 34-in.-diam break in Steam Line A. In the unaffected steamline (SG B) the TSV was closed, and the TBV system operated as designed. All of the other systems also operated as designed except for the parametric cases. The significant features and initial conditions for the MSLB calculations were: (1). Full reactor power. (2). Nominal temperatures and pressures in primary/secondary. (3). Decay heat - 1.0 times ANS standard. (4). Reactor and turbine trips coincident with MSLB. (5). Operator fails to isolate feedwater to both SGs until 600 s. (6). Operator restores unaffected SG (SG B) at 900 s. (7). RCPs restarted after 42 K subcooling reached. (8). HPI throttled to maintain 42 ± 12.5 K subcooling.

Case 2 was identical to the base case except the EFW system did not actuate as designed because of a modeling error in the input deck. In case 3, in addition to the EFW system's failing to actuate as designed, the RCPs never restarted after the subcooling margin was reached because of input deck errors. In case 4, the MFW pump was tripped at 0.5 s and the subcooling monitor for restarting the RCPs was moved from the hot leg to the top of the core, the proper location. Although these parametric cases were not specified, they are still useful calculations because they give other possible scenarios that possibly could occur during a MSLB transient and are useful for extrapolation activities.

Table 4.13 gives the sequence of events for the base case calculation. Figures 4.36 through 4.38 summarize the results of the base case calculation. The main forcing function for the overcooling was a specified delay of the operator in isolating the affected steam generator coupled with a delay in throttling the HPI flow. The base case analyzed had all plant protection and control systems operate as designed. The minimum downcomer fluid temperature calculated for the base case was 405 K. Repressurization of the primary system to the PORV set point was calculated for the base case following an initial depressurization to 3.5 MPa.

In the first two parametric cases (cases 2 and 3), input and modeling errors prevented the EFW system from operating as designed. Case 3 had an additional input error that prevented the RCPs from restarting once adequate subcooling was achieved. For these parametric cases, the downcomer fluid temperatures were considerably higher than the base case

Table 4.13 Main steamline break (base case) sequence of events

Event	Time(s)
MSLB loop A steam line	0.0
Turbine and reactor trip; TSV loop B closes	0.5
TBV loop B opens (set point 7.063 MPa)	5.0
HPI initiation (set point 10.44 MPa)	21.2
SG A low-level limit reached; loop A SG EFW flow initiated	29.4
TBV loop B closes	39.9
MFW pump trip on low suction pressure	47.8
SG B low-level limit reached; loop B EFW flow initiated	48.8
RCPs trip (30 s after HPI initiation)	51.2
Condensate booster pump trip (low suction pressure)	53.9
SG B level at 50%; loop B EFW valve closed	346.7
RCPs (A1, B1) restart (42 K subcooling)	526.0
Loop A, B accumulator set points reached (set point 4.17 MPa)	530.9
Loop A, B accumulators off	537.9
SG A, B isolated EFW pump and hotwell pump tripped off	600.0
SG B restored	900.0
PORV set point reached (set point 16.9 MPa) PORV opens and closes for remainder of calculation	4678-7200.0
TBV loop B opens (set point 7.063 MPa) TBV opens and closes for remainder of calculation to maintain setpoint pressure	5462-7200.0
SG B level drops below 50% operating range EFW initiated EFW pump on/off for remainder of calculation to maintain level at 50%	6121-7200.0
Calculation terminated	7200.0

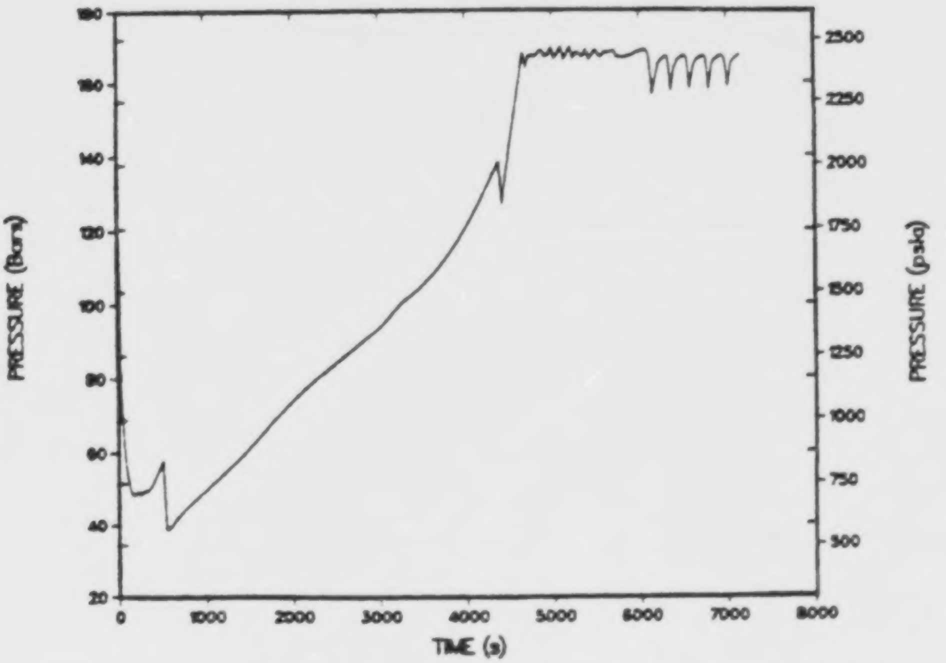

Figure 4.36 Pressurizer pressure, main steam line break - base case.

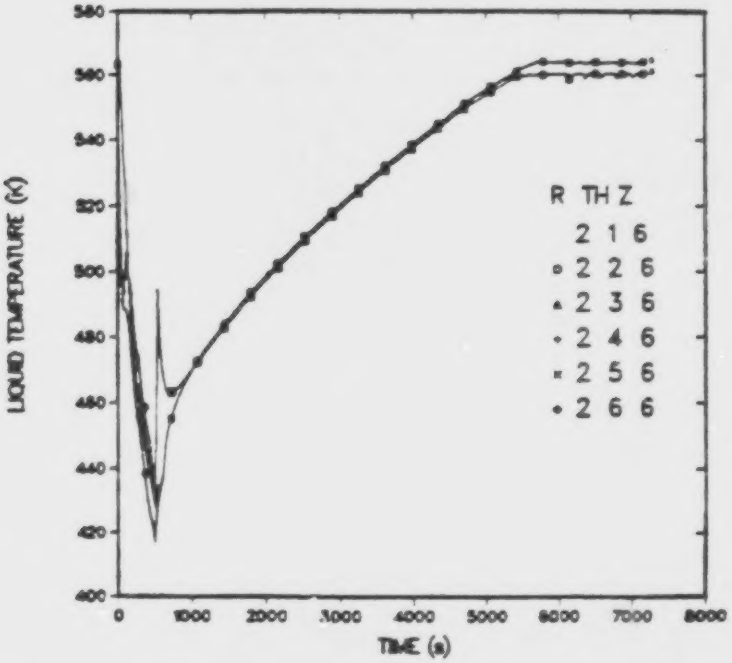

Figure 4.37 Downcomer liquid temperatures, main steam line break - base case.

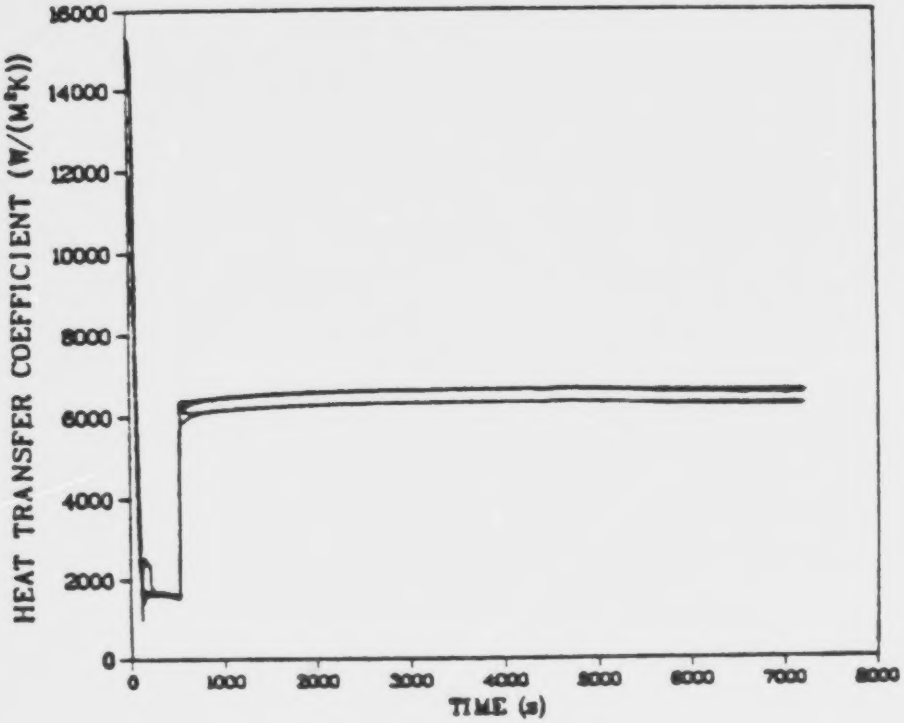

Figure 4.38 Heat-transfer coefficients, main steam line break - base case.

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(case 2 475 K; case 3 450 K); thus greater margins against PTS were calculated.

In the last parametric case (case 4), other changes were made to the model to reflect information provided by Duke Power after the base case was run. In this case, the subcooling monitor was corrected and the MFW pump was tripped at 0.5 s. This case was run to 2100 s, and the minimum downcomer fluid temperature was 420 K. None of the changes incorporated into case 4 resulted in significant differences from the base case.

4.3.2.2 PORV LOCA Transient

The PORV LOCA sequence calculated by LANL is transient 39 (LANL5) in Table 4.25. This case is the same as the INEL PORV small hot leg break, except for the tripping of the RCPs by the operator in the LANL case.

The PORV was assumed to open at transient initiation and remain stuck open for the remainder of the accident sequence. This event was followed by the reactor and turbine trips from full power. In addition to the PORV failure, the ICS failed to run back the main feedwater. As a result, the steam generators continued to fill until the MFW pumps were tripped on high SG level signal. To achieve this overfill, the high discharge and low suction pressure trips of the MFW pumps were overridden. The PORV failure, MFW trip override, and the ICS failure were the only assumed system-related failures. The RCPs tripped 30 s after HPI initiation, and this was the only specified operator action.

Table 4.14 gives the sequence of events for the calculation. Figures 4.39

Table 4.14 PORV LOCA Sequence of Events

Event	Time(s)
PORV opens	0.0
Turbine and reactor trip	0.5
Turbine stop valves close	0.5
Secondary-side heater and heater drain trip	1.1
Condenser feed from turbine trip	1.6
TBV-loop A opens for first time	4.4
TBV-loop B opens for first time	4.7
Condensate-booster pump trips on low suction pressure	11.0
TBV-loops A and B open/close	16.2
HPI actuation on low primary system pressure	70.3
TBV-loops A and B open/close	71.1
RCPs trip 30 s after HPI actuation	100.3
MFW realigned to SG upper headers	100.3
MFCV override trip	100.3
MFW pump trip on high SG level (loop A)	250.3
TBV-loop B opens for last time	330.0
Minimum primary pressure (7.2 MPa) attained	550.0
Pressurizer water-solid	600.0
Maximum primary repressurization (11.5 MPa)	850.0
End of calculation	1000.0

through 4.41 summarize the results of the calculation and extrapolation to 7200 s.

TRAC calculated a minimum vessel downcomer liquid temperature of 528 K between 600 s and 700 s into the transient. The primary system was calculated to repressurize to 11.5 MPa after 800 s. After recovering briefly between 800 s and 1000 s, the downcomer temperature was extrapolated to 527 (489°F) at 7200 s. Pressure remained at 11.5 MPa to the end of the transient.

4.3.2.3 Failure of Two Turbine Bypass Valves at Full Power

The LANL calculations of two turbine bypass valve (TBV) failures at full power include a base case (5A) and two parametric cases (5B and 5C). These sequences appear in Table 4.25 as transients 40 (LANL6), 41 (LANL7), and 42 (LANL8).

The significant features of the 2-TBV failure transient base case (5A) are:

(1). Reactor and turbine trips cause the TBVs to open.

(2). Failure of one bank of TBVs to close causes a secondary-side depressurization through the affected loop.

(3). The ICS fails to runback to the affected steam generator.

(4). Failure of the SG liquid level control in the affected loop follows initiation of EFW.

(5). The operator does not restart the RCPs.

(6). The operator does not throttle the HPI.

(7). The operator does not isolate/control feedwater.

Two parametric cases were also calculated. The steam generator liquid-level control in the affected loop operates correctly in case 1 (5B).

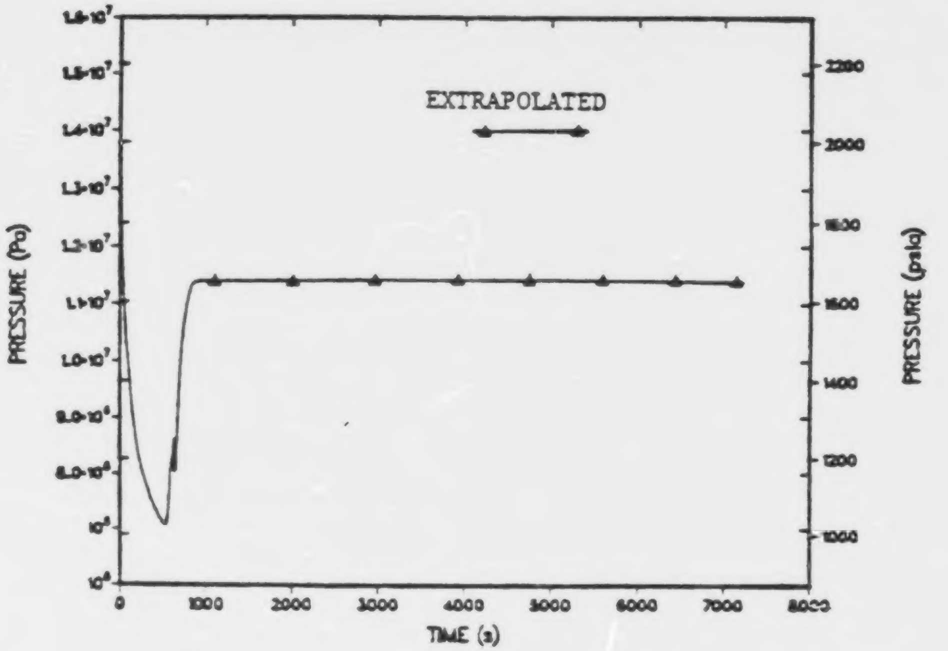

Figure 4.39 PORV LOCA extrapolated system pressure.

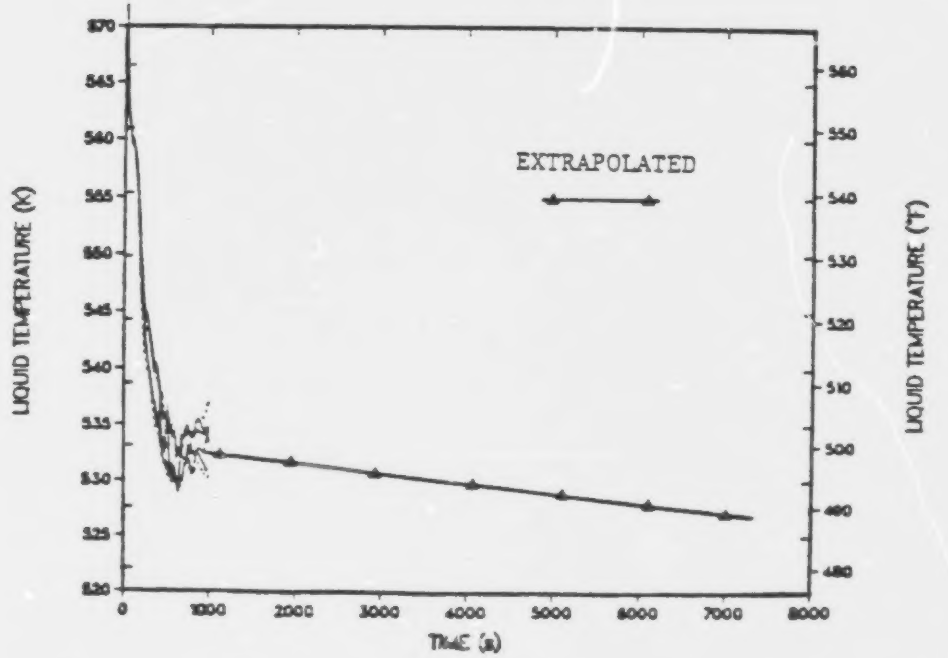

Figure 4.40 PORV LOCA extrapolated downcomer liquid temperature.

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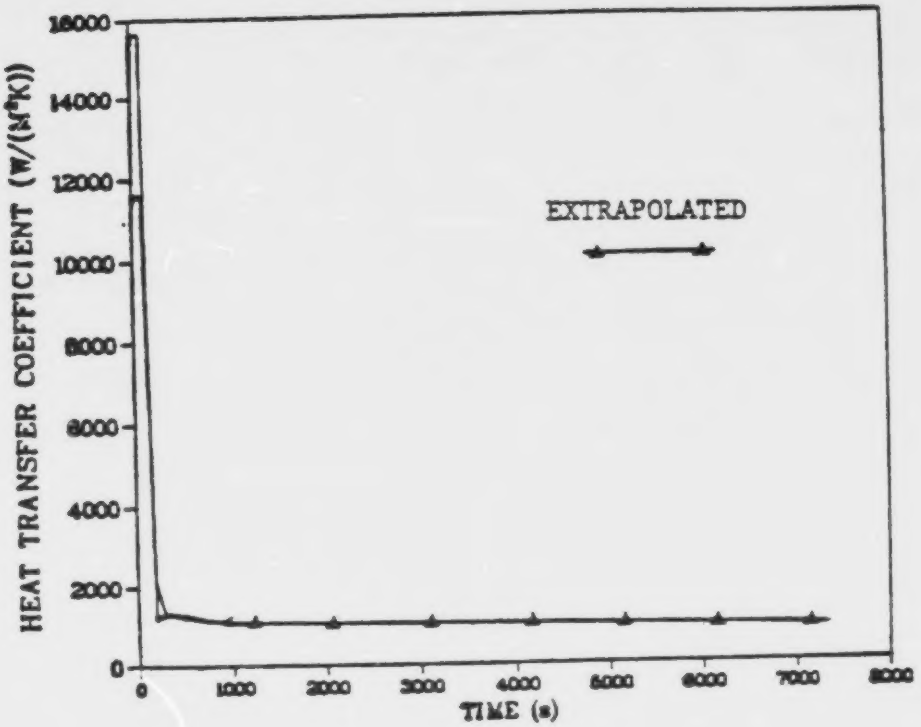

Figure 4.41 PORV LOCA extrapolated downcomer heat-transfer coefficient.

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The steam generator liquid level control also operates correctly in case 2 (5C), and operator actions to restart the RCPs and throttle the HPI are permitted if the primary system subcooling-monitor trip points are exceeded.

Table 4.15 gives the sequence of events for the base case. The parametric cases follow this exact sequence to the starting of the EFW pump at 209 s. In parametric case 1, the closure of the EFW control valve on SGA (290 s) speeds the closure of the EFW valve to SGB (460 s) and opening of the PORV (975 s). In parametric case 2, the closure of the EFW valve to SGA (290 s) brings about RCP restart at 383 s, closure of the EFW valve to SGB (395 s), and throttling of HPI (485 s).

Figure 4.42 gives the calculated and extrapolated pressures for the base and parametric cases. Figures 4.43 through 4.45 gives the calculated and extrapolated temperatures for the base and parametric cases. Figure 4.46 gives the calculated and extrapolated heat transfer coefficient for the base case and parametric Case 1. After restart of 1 RCP/loop in parametric case 2, a final heat transfer coefficient of 7500 w/m^2 K is obtained. The failure to isolate main and emergency feedwater to the affected steam generator resulted in very low temperatures for all three cases at 7200 s. The base case was extrapolated to 365 K. Parametric cases 1 and 2 were extrapolated to 440 K and 430 K respectively.

4.3.2.4 Failure of Four Turbine Bypass Valves at Full Power

The LANL calculations of four turbine bypass valves (TBV) at full power, like the two-TBV failure cases (Section 4.3.2.3), also include a base case

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Table 4.15 2-TBV failure full power sequence of events (base case)

Event	Time(s)
Turbine and reactor trip	0.5
Turbine stop valves close	0.5
TBV loop A opens (fails to reseat thereafter)	4.1
TBV loop B opens	4.3
MFW pump trip on high SG A liquid level	60.7
HPI started following trip on low pressure	153.1
RCPs trip on 30-s delay after HPI actuation	183.0
Feedwater realignment	183.0
Main flow control valves overriding trips	183.0
EFW flow initiated	209.1
Loop-B EFW valve shut on high SG liquid level	460.8
PORV opens	1036.7

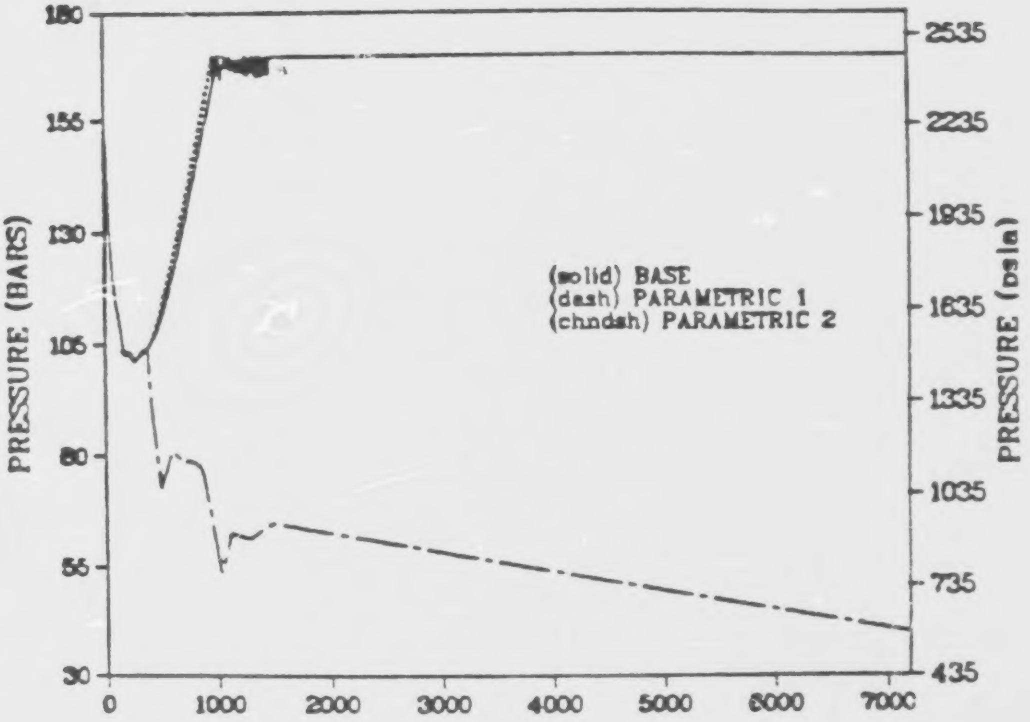

Figure 4.42 Pressurizer pressure histories for 2 TBV failures (case 5A-base; case 5B-parametric; case 5C-parametric 2).

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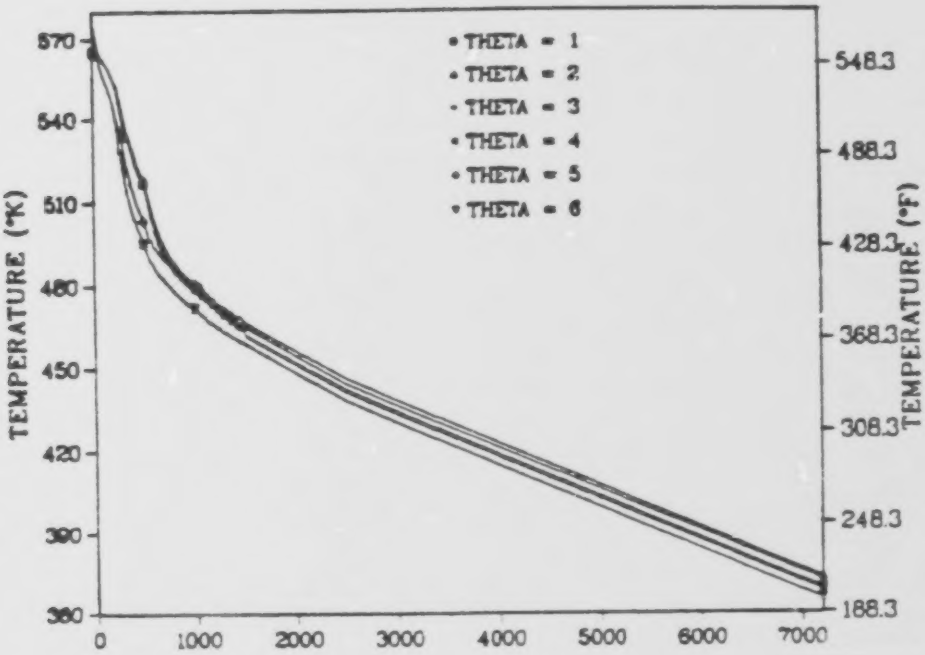

Figure 4.43 Downcomer liquid temperatures for case 5A.

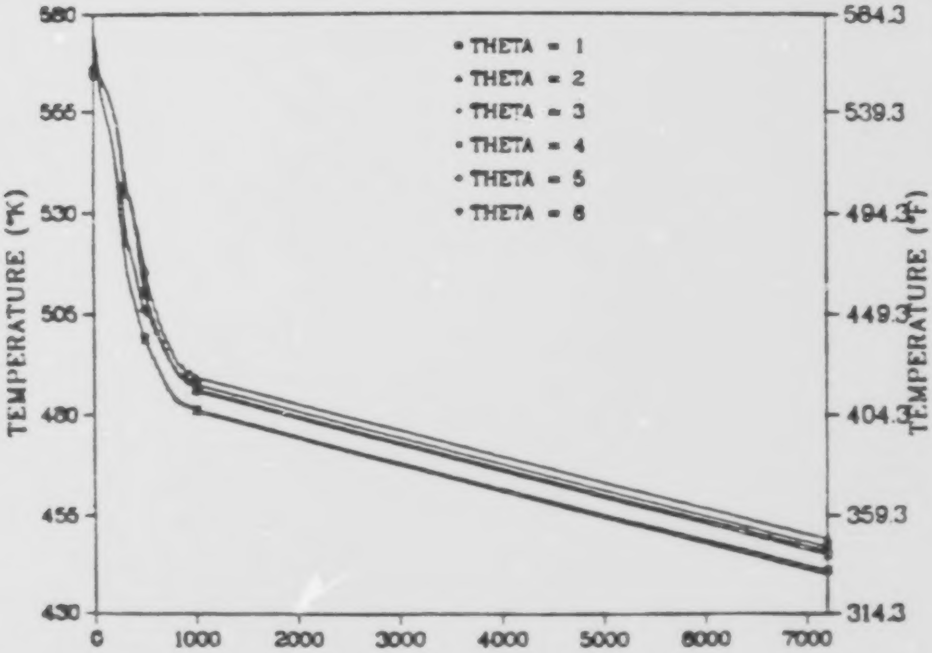

Figure 4.44 Downcomer liquid temperatures for case 5B.

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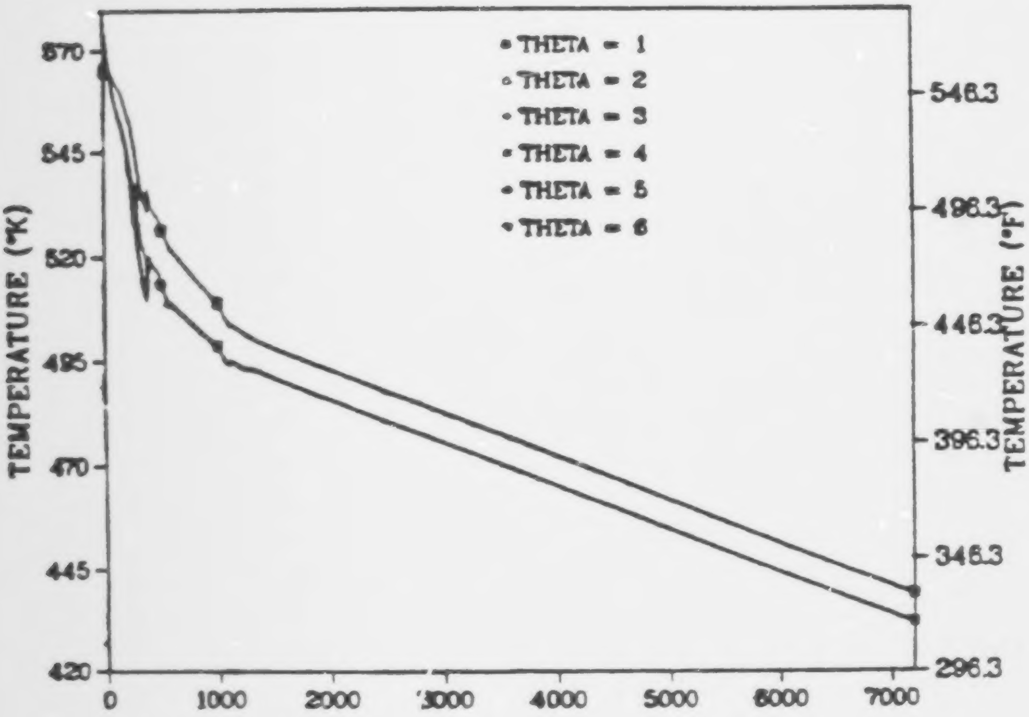

Figure 4.45 Downcomer liquid temperatures for case 5C.

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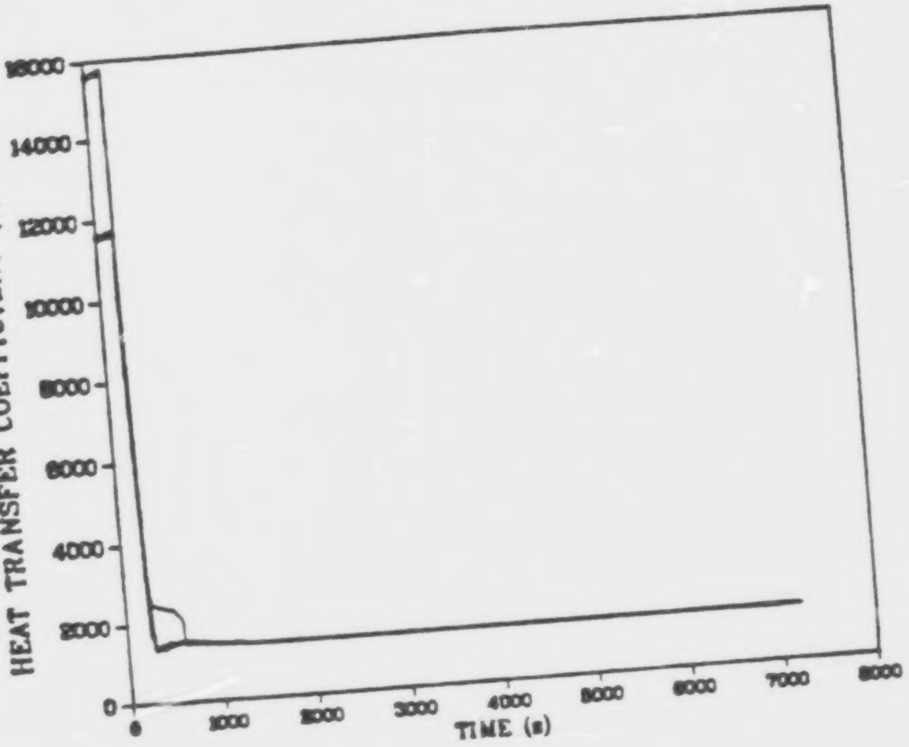

Figure 4.46 Heat-transfer coefficients for case 5A.

(6A) and two parametric cases (6B and 6C). These sequences appear in Table 4.25 as transients 43 (LANL9), 44 (LANL10), and 45 (LANL11) respectively. The base and parametric case descriptions are the same as for the two-TBV failure cases except that all four TBVs (two on each steam line) fail to reseat after opening.

Table 4.16 gives the sequence of events for the base case. Parametric case 1 behavior differs from the base case only in the proper operation of the EFW controls. Parametric case 2 is also similar to the base case until the HPI is throttled (421 s) and 1 RCP/loop is restarted (517 s). The increased cooling due to RCP operation reduces primary pressure so that accumulator injection occurs at 566 s. Figure 4.47 gives the calculated and extrapolated pressures for the base and parametric cases. Figures 4.48 through 4.50 likewise give calculated and extrapolated temperatures. Figure 4.51 gives the heat transfer coefficient for the base case and parametric case 1. After restart of the RCPs, the heat transfer coefficient for parametric case 2 assumes a final value of 7500 w/m^2 K. As in the two-TBV failure case, the failure to isolate main and emergency feedwater to both steam generators resulted in very low temperatures. Minimum temperatures of 350 K, 465 K, and 350 K respectively were extrapolated for the base and parametric cases.

4.3.2.5 Small Break LOCA Two-In. Hot Leg Break

The 2-in.-diam hot leg break LOCA calculated by LANL models the same conditions as the INEL pressurizer surge line small (2-in.-diam) break transient. The LANL calculation is transient 47 (LANL 13) in Table 4.25.

Table 4.16 4-TBV failure full-power sequence of events (base case)

Event	Time (s)
Turbine and reactor trip	0.5
Turbine stop valves close	0.5
Loop A TBV opens (fails to reseat thereafter)	4.1
Loop B TBV opens (fails to reseat thereafter)	4.3
HPI begins following trip on low system pressure	87.5
MFW pump trip off following high SG B liquid level	91.2
RCPs trip on 30-s delay after HPI actuation	117.4
Feedwater realignment trip	117.4
Main flow control valves trip	117.4
Emergency feedwater trip	117.4
PORV opens	1175.7

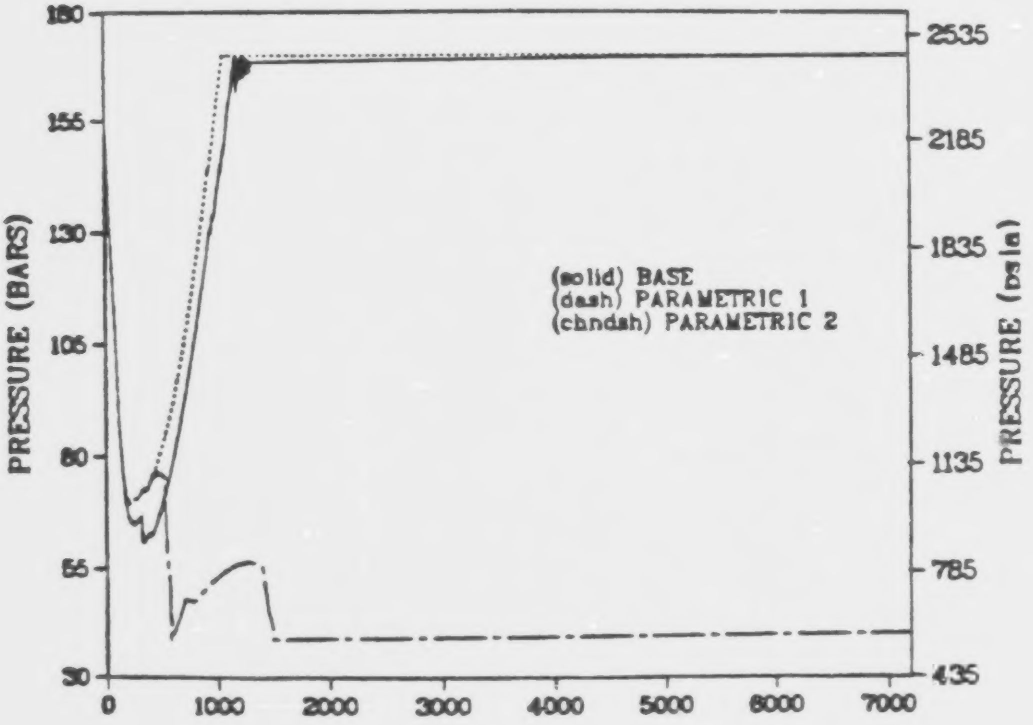

Figure 4.47 Pressurizer pressure histories for 4 TBV failures (case 6A-base; case 6B-parametirc 1; case 6C-parametric 2).

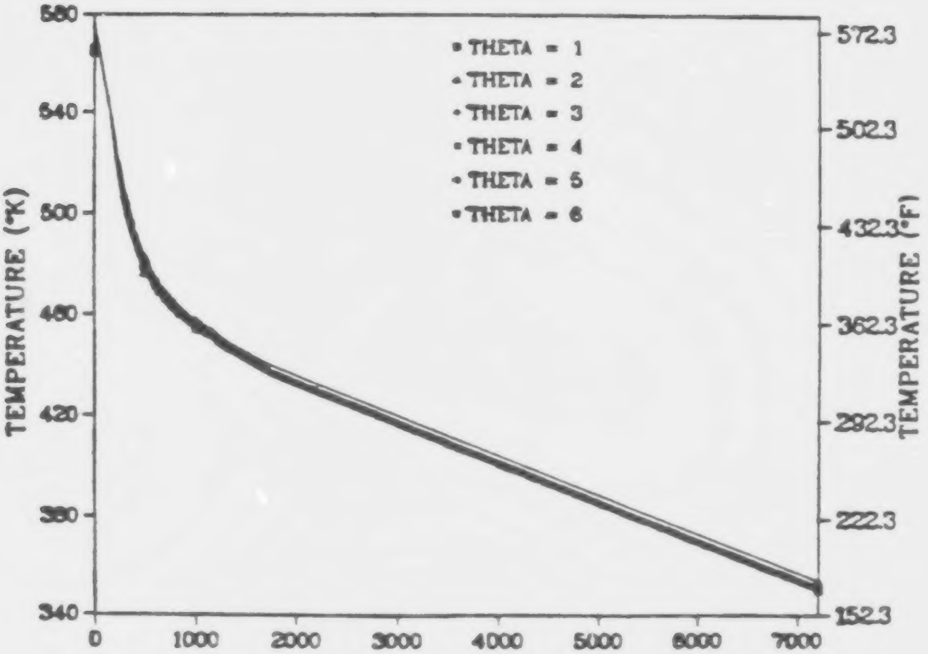

Figure 4.48 Downcomer liquid temperatures for case 6A.

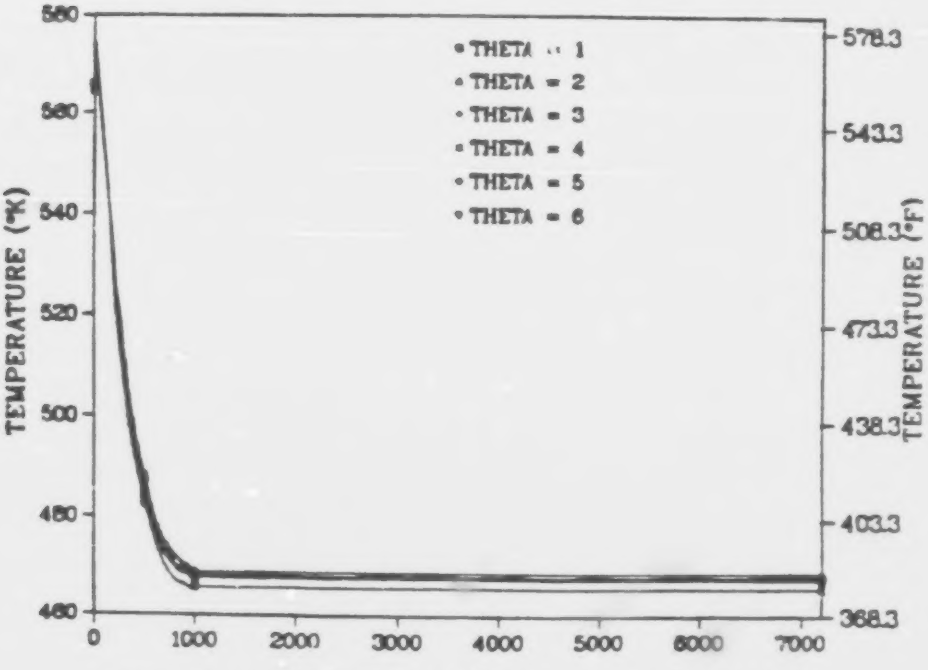

Figure 4.49 Downcomer liquid temperatures for case 6B.

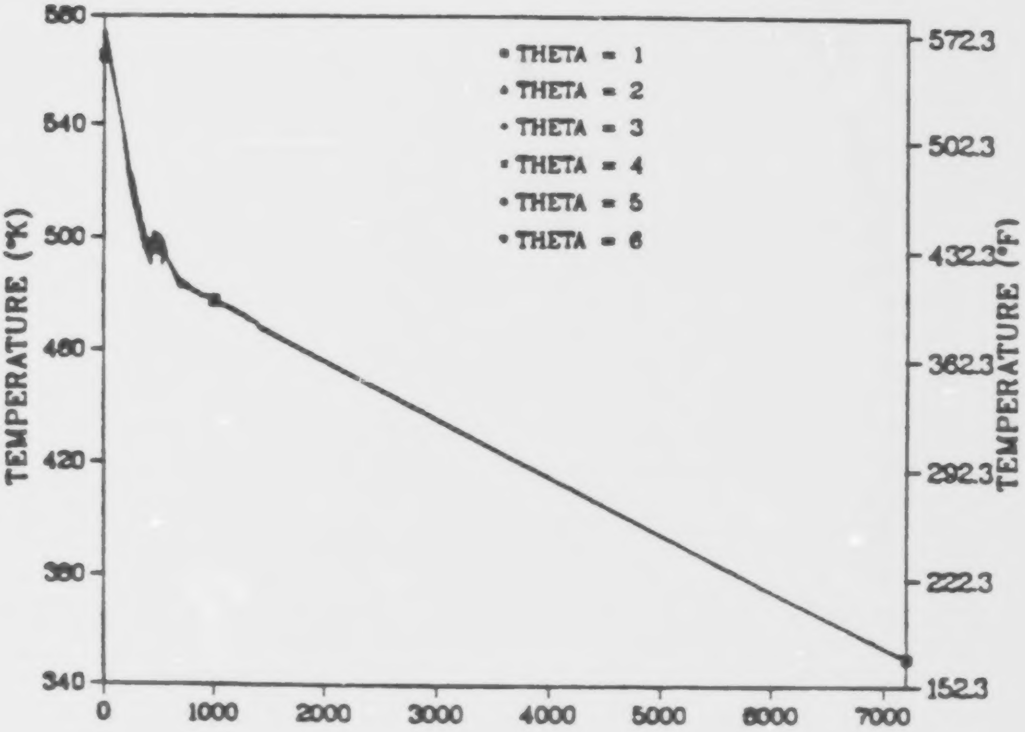

Figure 4.50 Downcomer liquid temperatures for case 6C.

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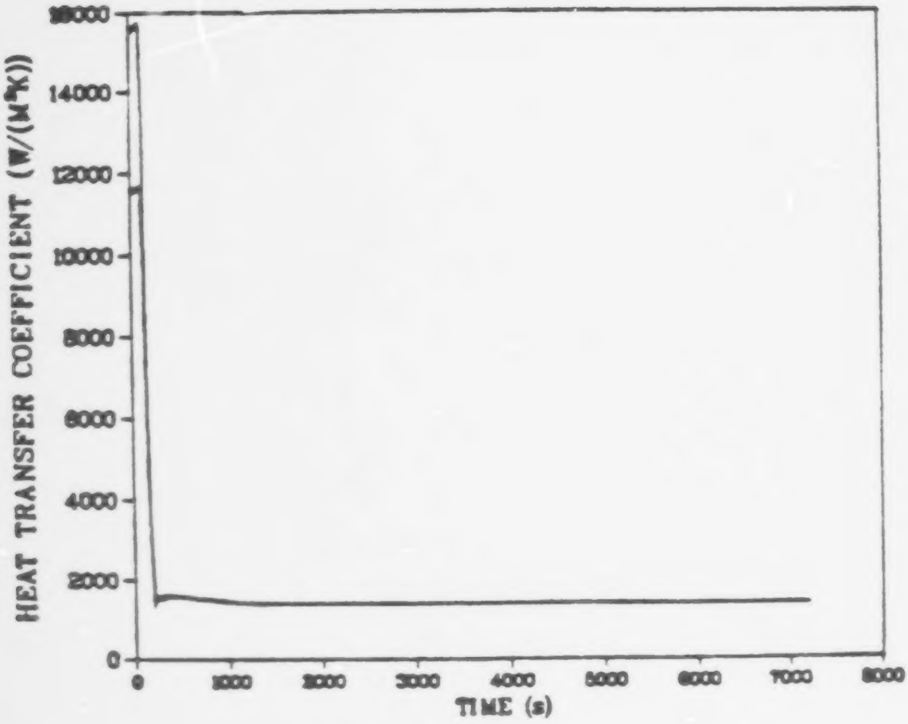

Figure 4.51 Heat-transfer coefficients for case 6A.

The sequence is initiated by a 2-in.-diam break in the surge line midway between the pressurizer and riser of the hot leg.

Following the initiation of the break, the reactor and turbine trip from full power. Reactor decay heat was specified as 1.0 times the ANS standard. For this transient calculation, the ICS and all key system components were assumed to function correctly. The only specified operator action was the RCP trip 30 s after HPI actuation. Two cases involving HPI throttling to system subcooling were investigated. One calculation investigated the effects of HPI throttling to 42 ± 12.5 K subcooling and the other investigated the effects of no HPI throttling. The throttled HPI case was not run because the subcooling margin was never achieved.

Table 4.17 gives the sequence of events for the calculation. Figures 4.52 through 4.54 present the results of the calculation and extrapolations with and without LPI operation assumed.

TRAC calculated a relative minimum in temperature at approximately 1000 s into the transient; the pressure at 1000 s was 6.2 MPa. The temperature and pressure were both decreasing at the end of the calculation at 3760 s and had values of 450 K and 2.1 MPa respectively.

The calculated minimum downcomer liquid temperatures never approached the current NDT value of Oconee-1 for two reasons; (a) vent valve flow mixing with the fluid in the downcomer region and (b) calculated loop oscillations. However, if the calculation were continued, the LPI system would actuate as a result of the depressurization. The addition of LPI would probably result in downcomer liquid temperatures approaching or exceeding the current NDT

Table 4.17 Hot leg break LOCA -- 2-in. break sequence of events

Event	Time (s)
Break opens	0.0
Reactor and turbine trips	0.5
TSVs close (both loops)	0.5
TBVs open (both loops)	4.2
HPI actuation	43.1
TBVs open/close (both loops)	51.0
RCPs trip	71.0
MFW realignment	73.1
TBVs open/close (both loops)	75.7
Vent valves open	100
ICS closes SUFCVs	300
Candy canes remain voided	500
Minimum downcomer temperature (470 K; 6.2 MPa)	750
Loop flow oscillations begin	1200
Accumulator injection begins	1750
End of calculation	3670

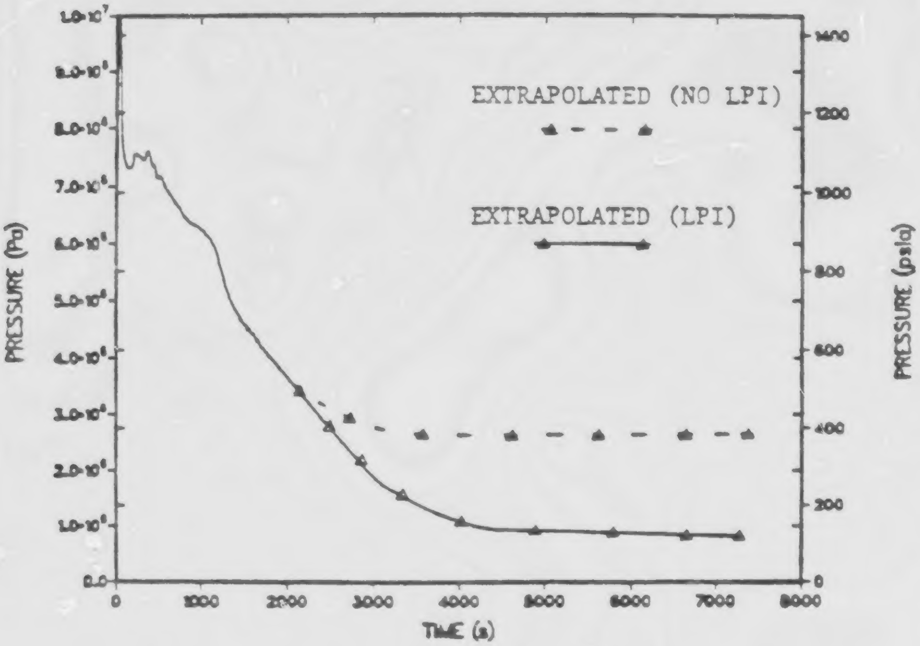

Figure 4.52 2-in.-diam SBLOCA extrapolated system pressure.

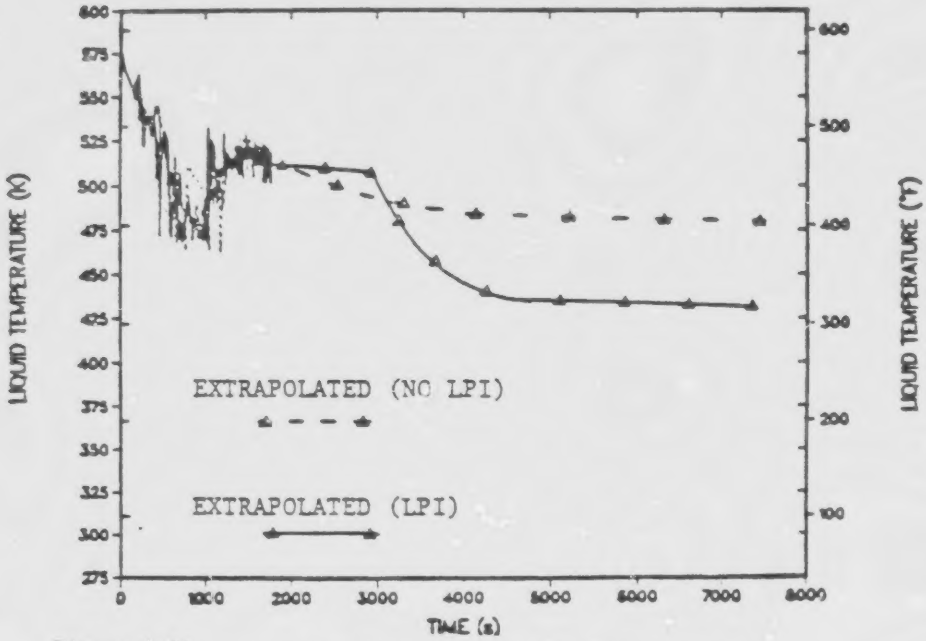

Figure 4.53 2-in.-diam SBLOCA extrapolated downcomer liquid temperature.

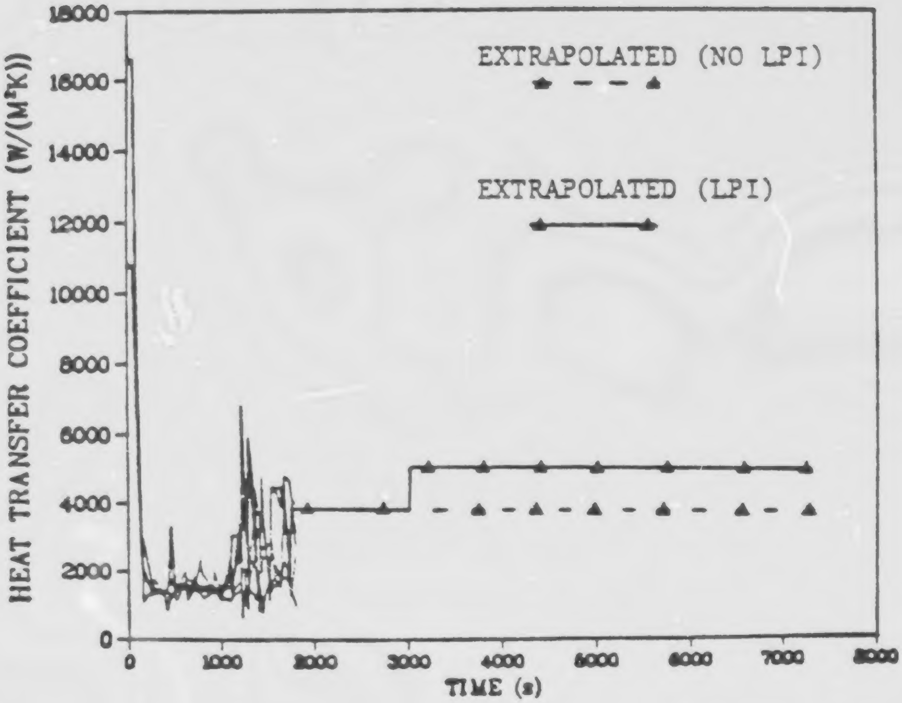

Figure 4.54 2-in.-diam SBLOCA extrapolated downcomer heat-transfer coefficient.

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of the Oconee-1 plant. However, this transient may not be important in terms of PTS because the primary system pressure will be quite low when the downcomer liquid temperature falls below the NDT limit.

4.3.2.6 Small Break LOCA 4-In. Hot Leg Break

The 4-in.-diam hot leg break LOCA calculated by LANL is transient 48 (LANL 14) in Table 4.25. This transient is initiated by a 4-in.-diam break in the surge line midway between the pressurizer and the riser portion of the hot leg.

Following the initiation of the break, the reactor and turbine trip from full power. Reactor decay heat was specified as 1.0 times the ANS standard. In this transient, the ICS and all key components are assumed to function correctly. The only specified operator action was the RCP trip 30 s after HPI actuation.

Table 4.18 gives the sequence of events for the calculation. Figures 4.55 through 4.57 summarize the results of the calculation and extrapolation to 7200 seconds. TRAC calculated a minimum downcomer liquid temperature of 350 K; the system pressure at this minimum temperature was 1.0 MPa.

Adequate fluid mixing between the vent-valve fluid and the cold leg (HPI) fluid in the downcomer at the cold leg junction maintained the downcomer liquid temperatures above 450 K. However, the actuation of the LPI system at 1240 s dropped the downcomer temperatures very rapidly and below the current NDT value (365 K) for the Oconee-1 plant. Even though the downcomer liquid temperatures were calculated to be below the current

Table 4.18 Small-break LOCA -- 4-in. Hot Leg Break

Event	Time (s)
Break opens	0.0
Turbine and reactor trip	0.5
Turbine stop valves close	0.5
Secondary side heater and heater drain trip	1.0
Condenser feed from turbine trip	1.5
TBV loop A opens for first time	4.4
TBV loop B opens for first time	4.8
HPI system actuation on low primary system pressure	16.8
TBV loops A and B open/close	117.4
RCPs trip 30 s after HPI actuation	46.8
MFW is realigned to SGs upper header	46.8
MFCV override trip	46.8
TBV loops A and B open/close	92.1
Candy canes void	125.0
ICS closes SUFCVs	300.0
Accumulator injection loop A (first time)	540.7
Accumulator injection loop B (first time)	541.0
Accumulator injections ceases	678.6
Accumulator injection loop A	726.1
Accumulator injection loop B	784.5
Accumulator injection ceases	828.7
Accumulator injection loop A	921.4
Accumulator injection loop B	925.7

Table 4.18 (continued)

Event	Time (s)
Accumulator injection ceases loop B	947.5
Accumulator injection ceases loop A	947.7
Accumulator injections (both loops)	1100.0
LPI system actuation on low primary system pressure	1236.0
End of calculation	1400.0

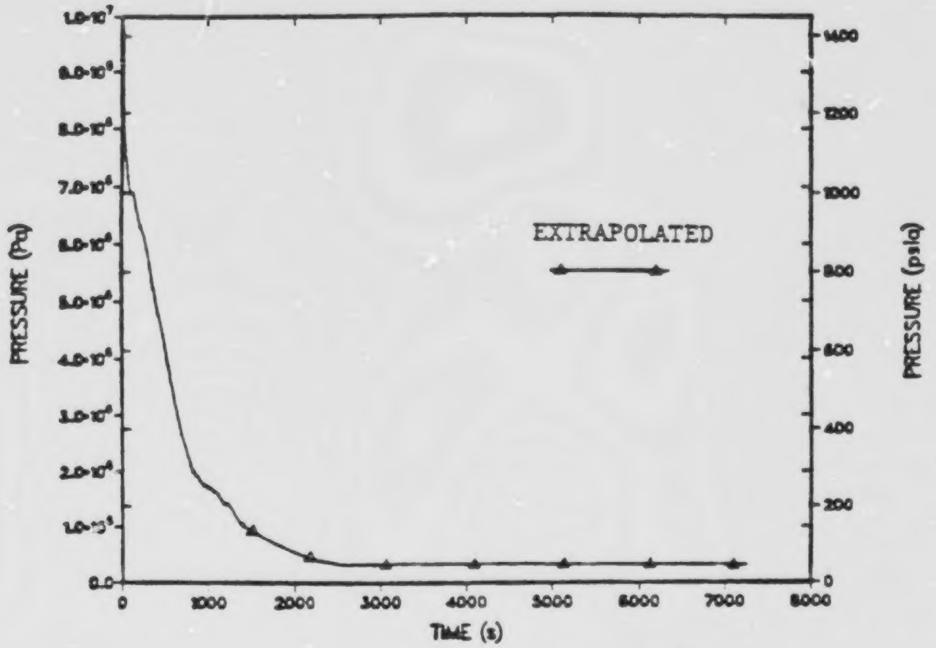

Figure 4.55 4-in.-diam SBLOCA extrapolated system pressure.

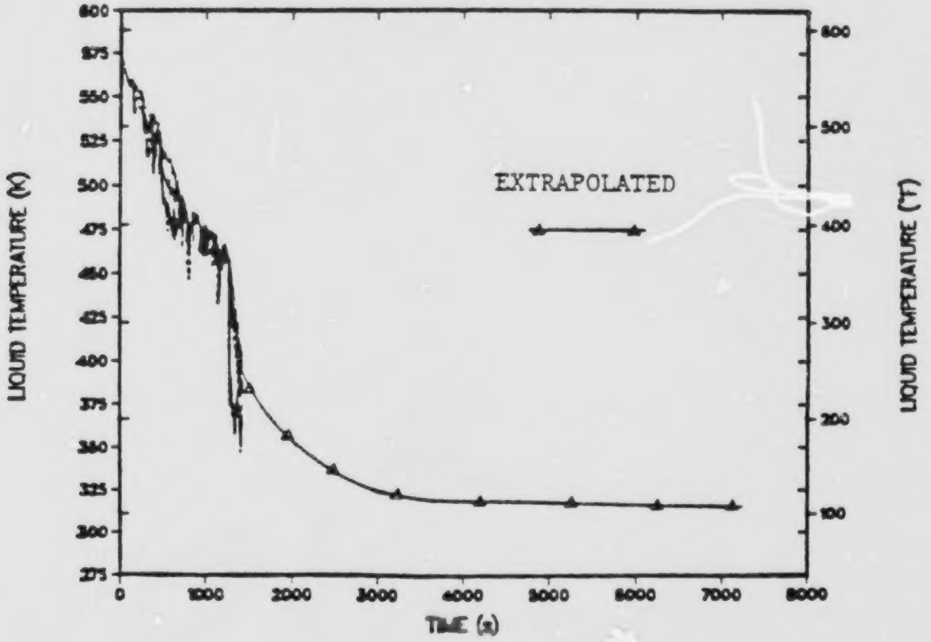

Figure 4.56 4-in.-diam SBLOCA extrapolated downcomer liquid temperature.

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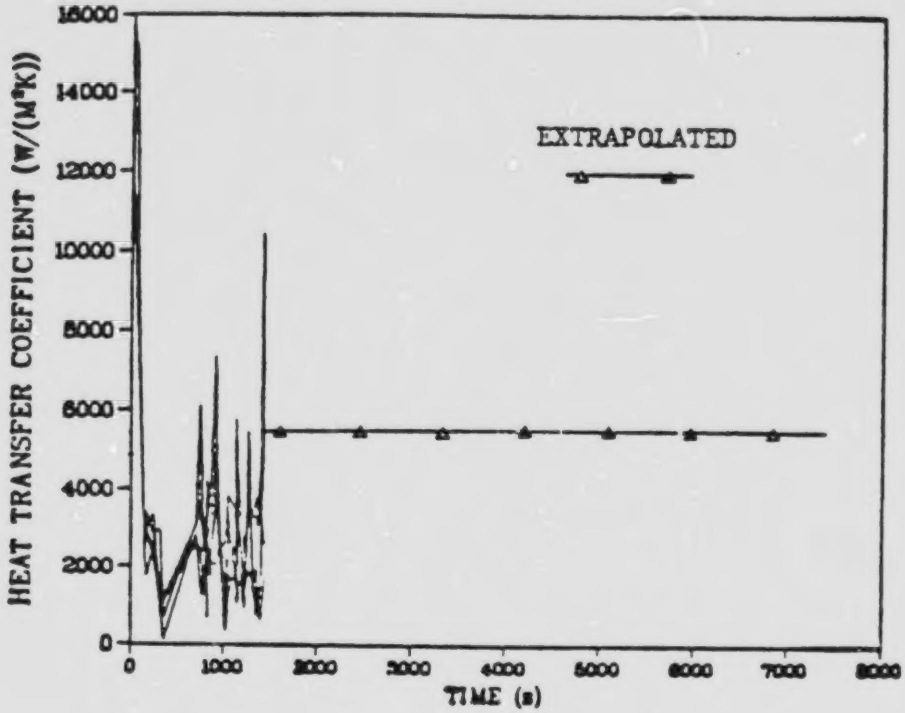

Figure 4.57 4-in.-diam SBLOCA extrapolated downcomer heat-transfer coefficient.

NDT Oconee-1, this calculation could not be considered a significant PTS transient because repressurization did not occur.

4.3.2.7 Steam Generator Dryout Followed by EFW Overfeed

The transient features steam generator dryout followed by emergency feedwater overfeed modeling the Rancho Seco transients as closely as possible on the Oconee-1 plant system. This sequence is transient 49 (LANL 15) on Table 4.25. The accident sequence began as a loss-of-main feedwater transient (MFW pumps trip). The EFW-control valves failed to open on demand but were subsequently manually opened by the operator. Also, the RCPs remained on during the transient, and the EFW to the SGs was not terminated until 4200 s. The primary system repressurization was limited to 13.8 MPa as a result of the operator's throttling the HPI system.

Table 4.19 gives the sequence of events for the calculation. Figures 4.58 through 4.60 summarize the calculation and extrapolations to 7200 s. At 4200 s the minimum downcomer fluid temperature of 452 K was reported. At the same instant, system pressure was 13.8 MPa. Both temperature and pressure were assumed to increase over the extrapolation period.

4.4 Evaluation of Flow Stratification Effects

The TRAC and RELAP5 codes cannot predict the formation of cold water plumes and similar flow stratification effects. Stratification occurs when cold and hot streams are brought together without sufficient turbulence to mix them. Cold HPI fluid, being denser, may form a layer at the bottom of the cold leg piping below the bulk of the warmer fluid in the cold leg. This layer or plume could run along the cold leg to the vessel and spill

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Table 4.19 Steam generator dryout followed by EFW overfeed (Rancho Seco-type) transient sequence of events

Event	Time (s)
MFW pumps trip, MFCVs and SUFCVs close	0.0
TSVs close (both loops)	0.0
Reactor trips on high pressure	4.4
TBVs actuated	4.6
PORV actuated	226.4
EFW initiated to both SGs	540.0
HPI actuated on low pressure	738.0
HPI throttled to limit repressurization	1255.0
EFW terminated to both SGs	4200.0
Minimum vessel downcomer liquid temperature (452 K) calculated	4200.0
Calculation terminated	4300.0

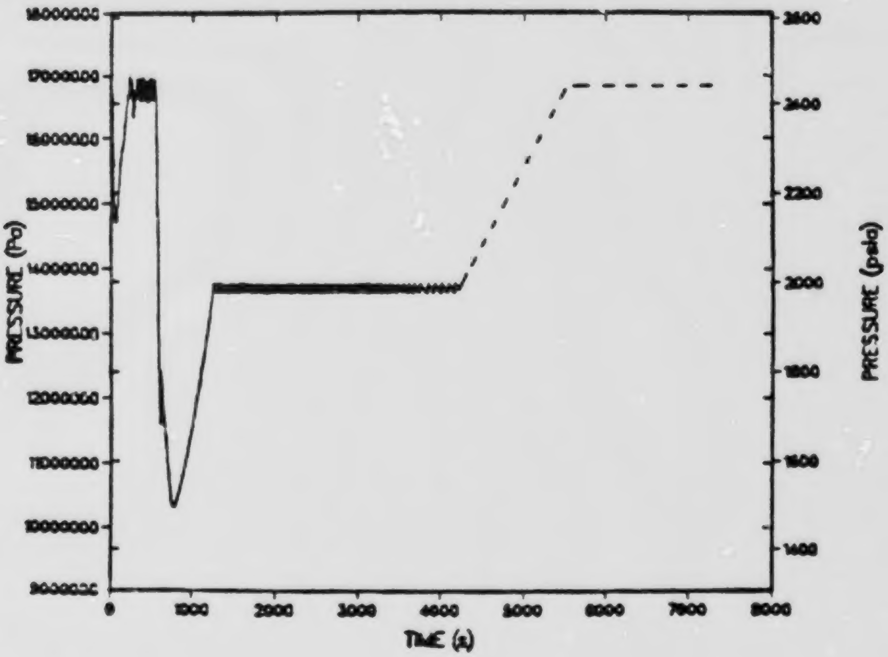

Figure 4.58 Rancho Seco-type transient extrapolated system pressure.

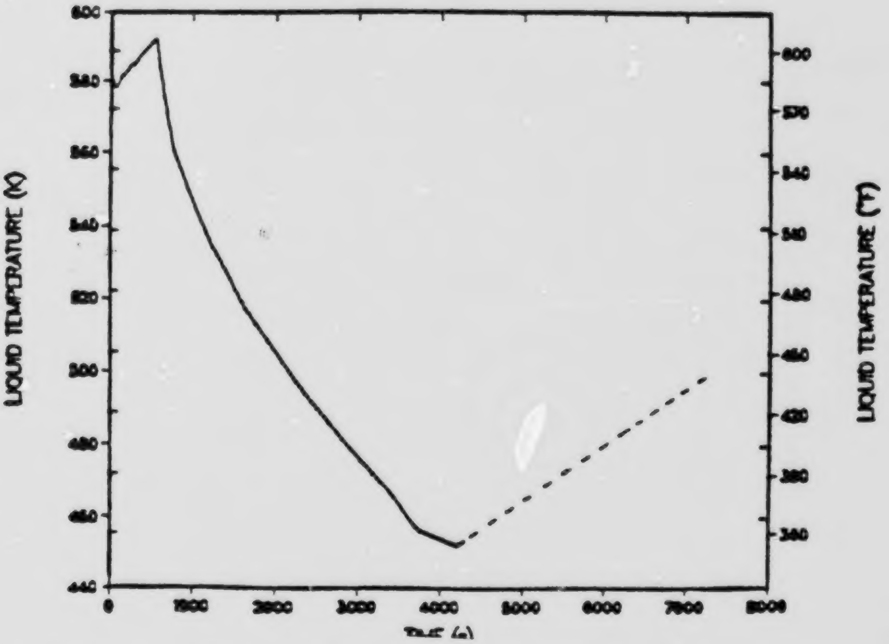

Figure 4.59 Rancho Seco-type transient extrapolated downcomer liquid temperature.

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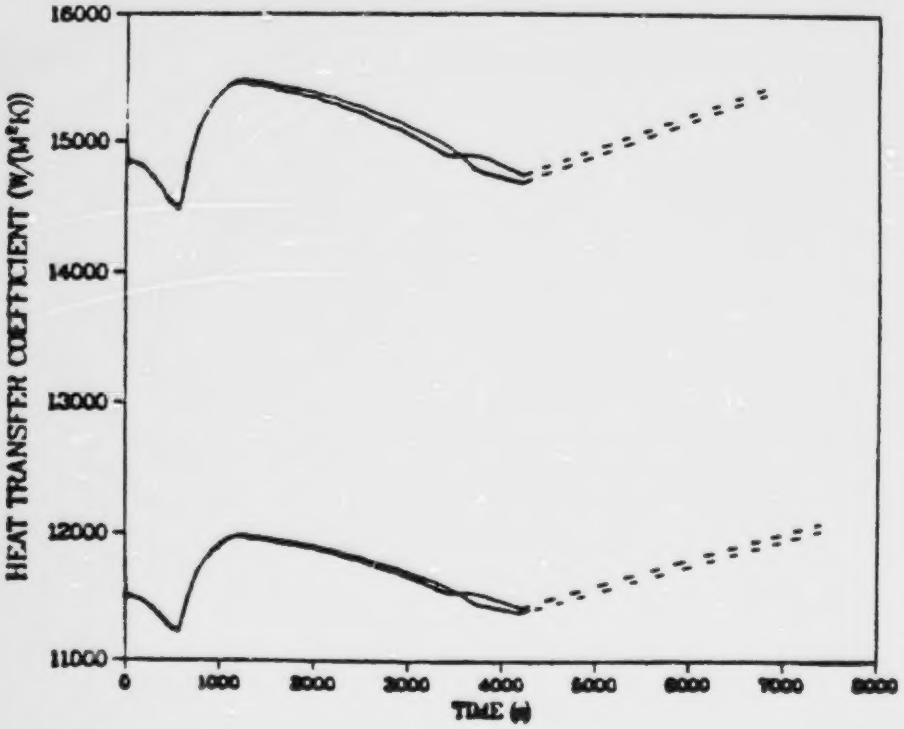

Figure 4.60 Rancho Seco-type transient extrapolated downcomer heat-transfer coefficients.

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down along the downcomer wall, producing localized cold temperatures on the downcomer which could greatly impact the fracture probability for the vessel. Factors affecting plume development include HPI nozzle and cold leg geometry, cold leg bulk flow and temperature conditions, vent valve flow rates and temperatures, and downcomer geometry.

This section summarizes a review by T.G. Theophanous,[4.3] Purdue University, of the INEL- and LANL-calculated transients for flow stratification effects. The approach of the review included the use of screening criteria based on empirical and theoretical mixing models to identify the cases where flow stratification may occur. Suspect cases were subjected to detailed mixing calculations to determine plume extent and temperature. The mixing model was benchmarked against experiments in a half-scale test facility set up for the Oconee-1 geometry.

The screening criteria eliminated all but seven of the detailed calculations from consideration of stratified flow. The INEL cases for which no stratification was predicted were: revised main steam line break, stuck open PORV, steam generator overfeed, maximum sustainable overfeed, and steam generator tube rupture. LANL cases included TBV cases 5B, 5C, 6A, 6b, and 6C; the main steam line break cases; PORV LOCA; and the steam generator dryout followed by EFW overfill cases.

For the seven remaining transients, it was determined that the TRAC and RELAP calculated bulk temperatures were adequate. The reasons are summarized below.

The LANL 4-in. hot leg break transient exhibited very low pressure when stratification would prevail (after 300 s). With this low pressure and two-phase flow conditions in the downcomer, the transient was not a PTS concern.

The INEL RC pump suction break also had very rapid depressurization. An asymmetric stagnation condition prevailed with vent valve flows equal to or greater than the unstagnated loop flows. Calculated plume temperatures stayed within 10 °C of the bulk temperature of the flowing loop. The strong vent valve flow will also break up the plumes.

The INEL pressurizer surge line break shows stagnation in loop A between 750 and 1250 s and in loop B between 1000 and 2500 s into the transient. Strong vent valve flow during these periods broke up the plumes. Also, planar plume mixing calculations showed restrictions to plume lateral growth such that the longitudinal welds were not affected.

The LANL two-TBV failure case (5A) is a high-pressure transient with one flowing loop (A) and one stagnant loop (B). Vent valve flow is strong, and is effective at mixing the downcomer. The plumes for the stagnant loop were found to be only 10 °C colder than the flowing loop.

The INEL original main steamline break case is also a high-pressure transient. Loop B stagnates between 250 and 950 s. Loop A stagnates between 200 and 350 s and again between 650 and 950 s. Vent valve flow is strong over these periods. Temperatures of 450 K (350 °F) were calculated at the stagnated cold leg nozzles. The strong vent valve flows prevented the growth of the plumes over the longitudinal welds.

The INEL four-TBV failure at hot standby case is a high-pressure transient in which there was not stagnation or vent valve flow. However, the low loop flows merited concern for plume development. The RELAP5 model predicted oscillating flows, which, if actually encountered, would generate excellent mixing. Even if plume development were significant, the excellent cold leg mixing characteristics and the general cooldown of the system would minimize the difference between bulk and plume temperatures.

The LANL 2-in. hot leg break case is a low-pressure transient. This case is similar to the corresponding INEL case.

In summary, the mixing geometry of the cold legs and the strong vent valve flows mitigated both the relative temperature decrease and the spread of plumes in high-pressure transients. In low-pressure transients the plumes could be colder, but the low pressures nullified the impact to risk of vessel failure.

4.5 Extrapolated Sequences

4.5.1 Methodology

4.5.1.1 General Approach

After an initial survey of the data resources and of the sequences identified for estimation, a five-step process was developed for the estimation of Oconee-1 pressure, temperature, and heat transfer coefficient profiles. This approach allowed logical reduction of the number of cases to be evaluated and derived the greatest benefit from the information in

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the TRAC and RELAP calculations. The approach is discussed further in Appendix C.

The first step involved the grouping of similar sequences. An evaluation of the TRAC and RELAP calculations for the effects from different operating states provided the criteria for assignment of sequences into groups. Besides providing grouping criteria, step 2 developed the parameters for the cooldown model used on occasion for this study. To assure correct interpretation of conditions during sequences, the appropriate parameters would be applied to the cooldown model to duplicate portions of sequences calculated by TRAC or RELAP. This validation effort took place in step 3.

In step 4, the pressure, temperature, and heat transfer coefficients were estimated. Temperature could be estimated either by piecewise application of TRAC and RELAP results or by calculation using the cooldown model. The method selection depended on the complexity of the sequence and the availability or absence of directly applicable data from the TRAC or RELAP calculations. Early portions of sequences where HPI, feedwater, and primary loop flows and downcomer temperatures vary rapidly were not directly applicable to the cooldown model, so piecewise use of TRAC and RELAP data was applied. Late in transients when the loop flows are more stable, a cooldown model was applied to predict the temperatures trends. The cooldown model is an energy balance which accounts for the heating and cooldown mechanisms present in the Oconee-1 system. The cooldown model was most useful where the end-state configurations in the requested sequence evaluations deviated significantly from those in the TRAC and

RELAP calculations. The derivation of the cooldown model is also included in Appendix C.

Pressure estimates were derived solely from observation of pressure trends in the TRAC and RELAP calculations. The ability of the high-head HPI system in the Oconee-1 plant to quickly repressurize the primary was reflected by the TRAC and RELAP5 calculations. The pressure behavior of the various calculations was sufficiently uniform to suggest that piecewise selection of TRAC and RELAP pressure curves was valid.

Heat transfer coefficients were based on piecewise selection of TRAC data. In general, the calculations predicted relatively constant values while the reactor cooling pumps (RCP) are running and step down to a lower but constant value after RCP trip and establishment of natural circulation. The heat transfer coefficients predicted by TRAC did not include correction for free convection effects. Therefore, the values were underpredicted for natural circulation flow conditions. This effect was compensated for in the fracture mechanics calculations by specifying a minimum heat transfer coefficient of 600 Btu/h $ft^{2o}F$ (3400 W/m^2K). Any lower values reported in this section or the appendix were superceded by the specified limit.

The completed estimations were documented in step 5. This documentation comprises Sections C.3 through C.8 of Appendix C.

4.5.1.2 Sequence Grouping

The overcooling sequence event trees formulated by ORNL possessed over 6.7 million end-states. Application of a 10^{-6}/yr screening limit on event frequency reduced the number of sequences requiring specific estimate to under 100. These were grouped into small sets by initiator, that is turbine bypass valve failures at full power, turbine bypass valve failures at hot standby, PORV-sized LOCA, feedwater overfeeds, steam generator tube ruptures, and main steam line breaks.

Some additional grouping of sequences based on component states' similarities further reduced the number of estimates to 97. However, grouping by component states (i.e., valve sticks open, ICS fails to runback feedwater, etc.) does not account for the thermal-hydraulic impact of such states. The actual impact will also depend on the function or failure of other systems and on feedback from thermal-hydraulic impact of such states. By observation and evaluation of similar events in the TRAC and RELAP calculations, the contributing events in a sequence were classified as dominant, minor, or inconsequential. Sequences with the same dominant events were grouped together for analysis. In later steps, the influence of minor events was evaluated where possible to check the consistency of the groupings. Some sequences were reassigned to other groups based on those checks.

The sequence groupings and the results of sequence extrapolation are presented by initiator in the following sections.

4.5.2 Turbine Bypass Valve Failures at Full Power

Evaluation of turbine bypass failures at full power concerned the 19 cases identified in Table 4.20. The table lists the specified conditions, which include various combinations of failed TBVs, feedwater system performance, primary pressure controls, and operator actions. These cases

Table 4.20 Full-power TBV failure case groupings

	Case (TBV-)																		
Condition*	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19
Secondary Pressure Controls																			
1 TBV open SGA	x	x	x	x	x	x	x		x	x			x	x	x				
1 TBV open SGA, SGB								x								x	x	x	x
1 TBV open SGA, 1 SRV open SGA											x	x							
Feedwater																			
MFW runback as designed	x		x					x	x	x	x	x	x					x	x
MFW overfeed SGA, high level trip				x										x			x		
MFW overfeed both SGs, HL trip						x	x												
MFWP trip at time = 0.0		x			x										x	x			
EFW control to level	x	x	x	x	x	x	x	x	x	x	x	x	x	x	x	x	x	x	x

Table 4.20 (continued)

Condition[*]	Case (TBV-) 1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19
Primary Pressure Controls																			
PORV block valve closed									x										
PORV operates properly	x	x	x	x	x	x	x	x			x	x		x	x	x	x		
PORF fails to reseat[**]										x			x					x	
SRV fails to reseat[**]													x	x	x				x
Operator Actions																			
RCP tripped at HPI + 30 s	x	x	x	x	x	x	x	x	x	x	x	x	x	x	x	x	x	x	x
No other actions	x	x					x	x	x		x	x	x	x	x	x	x	x	x
SG isolation at 20 min, SGB restored at 21 min			x	x	x	x				x									
HPI throttled later of 22 min or 50°F subcooling			x	x	x	x				x									
1 RCP/loop restart later of 22 min or 50°F subcooling			x	x	x	x				x									
Group number (TBVG-)	1	1	2	3	2	3	4	5	1	6	8	7	6	4	6	5	9	10	10

[*] Other conditions common to all cases: full power, decay heat 1.0 times ANS, all ECCS components operate as designed

[**] If challenged

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were consolidated into the 10 groups indicated at the bottom of Table 4.20. These groups represent transients 1 (TBVG1) through 10 (TBVG10) on Table 4.25. Figures 4.61 through 4.63 give the extrapolated pressures, temperatures, and heat transfer coefficients for groups 1, 2, 4, 6, and 7. These groups are representative of all groups as a whole. Groups 3 and 8 correspond closely to group 2, which is in the figure. Likewise, group 9 is similar to group 4, and groups 4 and 10 are similar to group 6 in their responses. Section C.4 of Appendix C provides detailed individual evaluations of these groups. The uniformity of temperature trends is in keeping with the mechanisms driving the cooldown. Group 2, the only group to experience reheating, does so upon the dryout of the affected steam generator after feedwater isolation. Groups 1, 6 and 7 all feature proper ICS feedwater control so that the steam generators are not flooded out. Therefore, these groups cluster together late in the sequences, although the equivalent of two stuck-open TBVs are in groups 6 and 7 vs only one in group 1. In TBV group 4, an ICS failure allows continued flow of main feedwater (from the hotwell and condensate pumps) to flood the affected steam generator(s). This is the case also for the LANL two-TBV failure and four-TBV failure cases. (The two-TBV base case (5A) values were therefore given to TBVG4.) This flooding produces the lowest extrapolated temperatures. The minimum temperatures for TBV groups 1, 2, 4, 6, and 7 are 420 K, 471 K, 365 K, 398 K, and 406 K respectively.

4.5.3 Turbine Bypass Valve Failures at Hot Standby

Evaluation of turbine bypass valve failures at hot standby involved the four cases identified in Table 4.21. The table lists the conditions modeled,

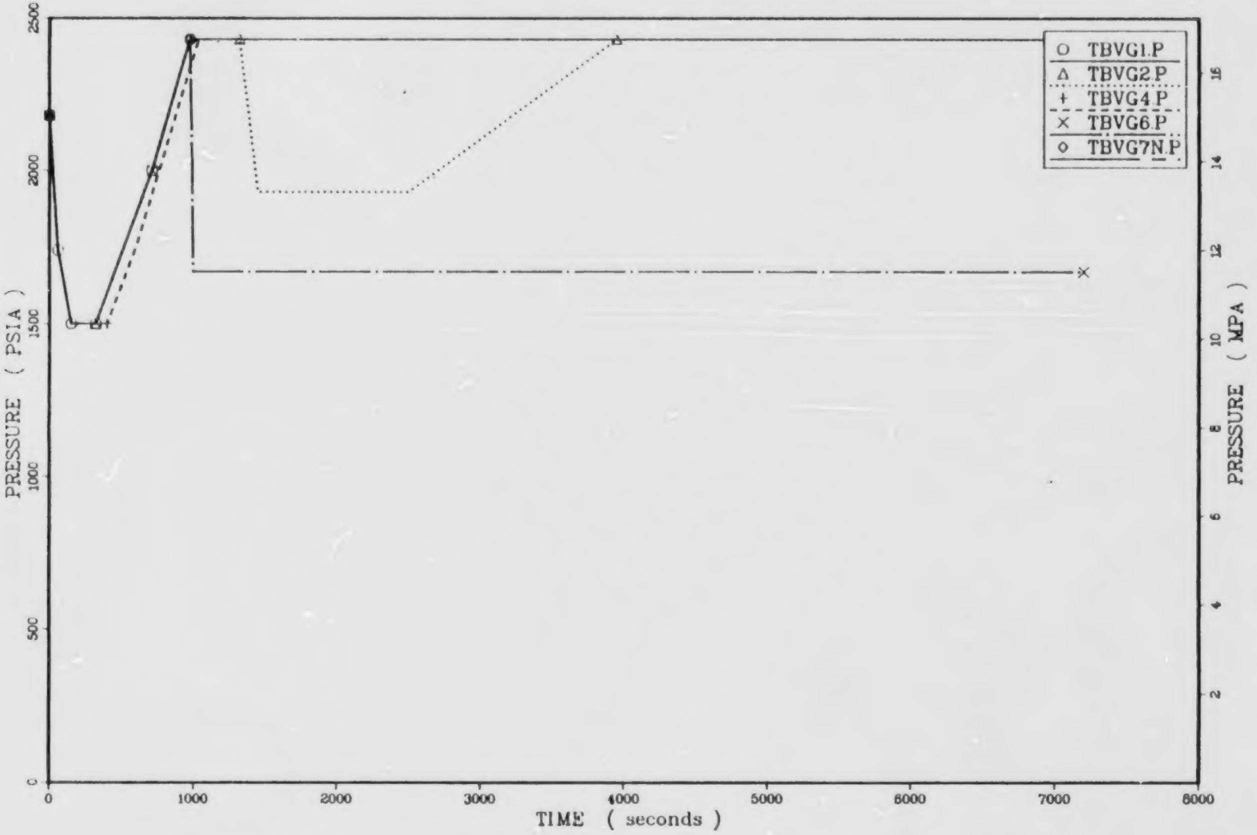

Figure 4.51 Extrapolated downcomer pressures for Oconee-1 TBV - full power cases.

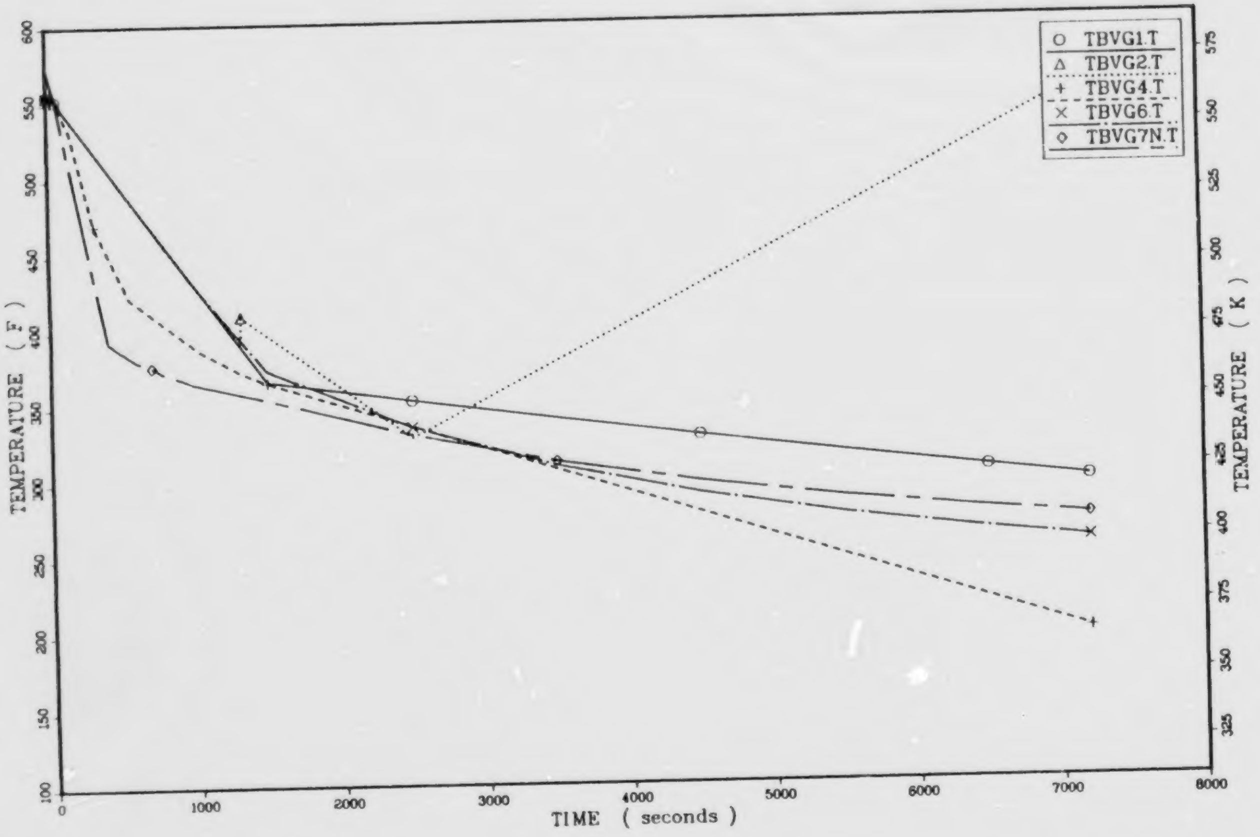

Figure 4.62 Extrapolated downcomer temperatures for Oconee-1 TBV - full power cases.

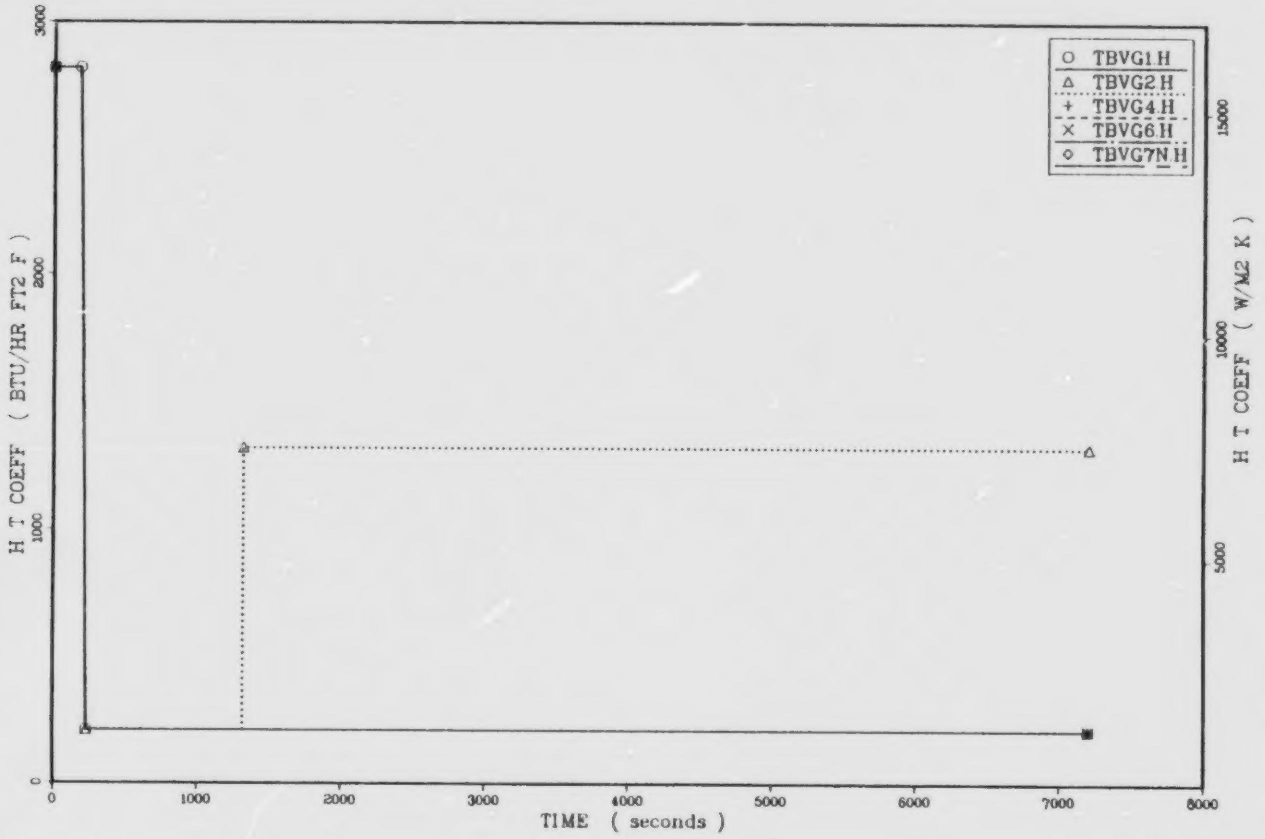

Figure 4.63 Extrapolated downcomer heat transfer coefficients for Oconee-1 TBV - full power cases.

Table 4.21 Hot standby TBV failure cases

	Case			
Conditions	1TBV (12 hr)	2TBV (12 hr)	1TBV (1 wk)	2TBV (1 wk)
Initial Conditions				
Hot standby	x	x	x	x
Decay heat levels				
a. 12 hrs = 15.75 MW	x	x		
b. 1 wk = 7.5 MW			x	x
Secondary Pressure Control				
1 TBV opens SGA	x		x	
2 TBVs open SGA		x		x
SGB TBV's operate as designed	x	x	x	x
SSRVs operate as designed	x	x	x	x
Feedwater Supply				
Feedwater controlled to appropriate level*	x	x	x	x
Primary Pressure Control				
PORV operates as designed	x	x	x	x
Operator Actions				
RCPs tripped at HPI + 30 s	x	x	x	x

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*Assumes post RCP trip SG level maintained by whatever feedwater source available.

which included one and two valve failures at power levels corresponding to 12 h and 1 wk after shutdown. These cases correspond to transients 24 (ZP1TBV), 25 (ZP2TBV), 33 (WITBV), and 34 (WZTBV) on Table 4.25. Figures 4.64 through 4.66 give the extrapolated pressures, temperatures, and heat transfer coefficients for the cases. The temperature trends show that decay heat effects are most notable at the end of the sequences where an offset of 14°C separates the corresponding one-valve and two-valve failures. The minimum temperatures for ZP1TBV, ZP2TBV, ZWITBV, and ZW2TBV are 385 K, 371 K, and 358 K. The failure to isolate the steam generators and throttle HPI led to lower temperatures than were obtained for the INEL four-TBV failure at hot standby calculation.

4.5.4 PORV-Sized LOCA Cases

Nine PORV-sized LOCA sequences identified for evaluation are listed in Table 4.22. These are reduced to five groups, as indicated at the bottom of Table 4.22. These groups are transients 11 (PSBG1) through 15 (PSBG5) in Table 4.25. Figures 4.67, 4.68, and 4.69 give the extrapolated pressures, temperatures, and heat transfer coefficients for groups 1, 3, 4, and 5. Group 2 is very similar to group 1. Detailed descriptions of the groups and the extrapolation procedure are available in Appendix C, Section C.6. The main differences between the sequences are due to the feedwater situation accompanying the primary break. The PORV-sized LOCA results in sufficient HPI flow so that nearly all of the decay heat can be absorbed by the primary, even early in the transient. Later, after the decay heat has died down, the HPI is able to cool the system somewhat. In this situation, little or no steam is generated in the secondaries. Therefore, both normal

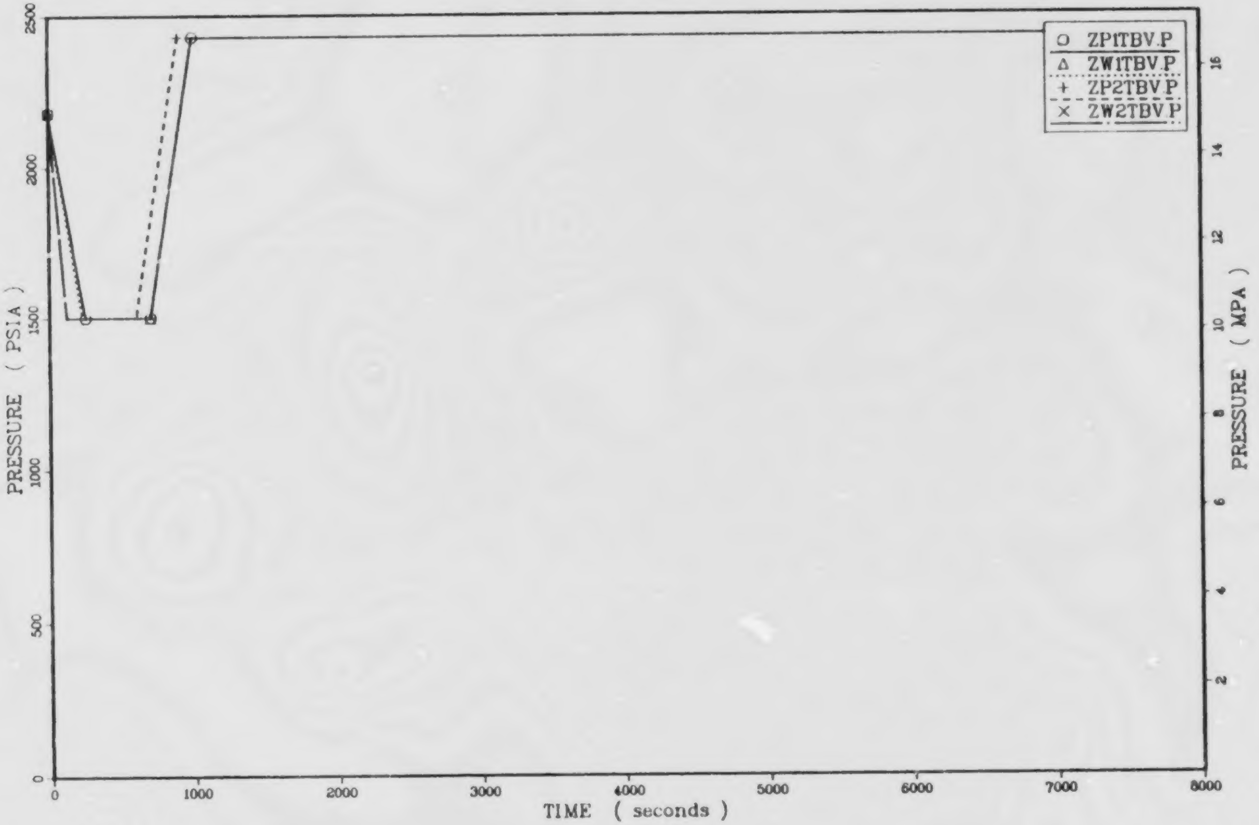

Figure 4.64 Extrapolated downcomer pressures for Oconee-1 TBV failure at hot stand-by cases.

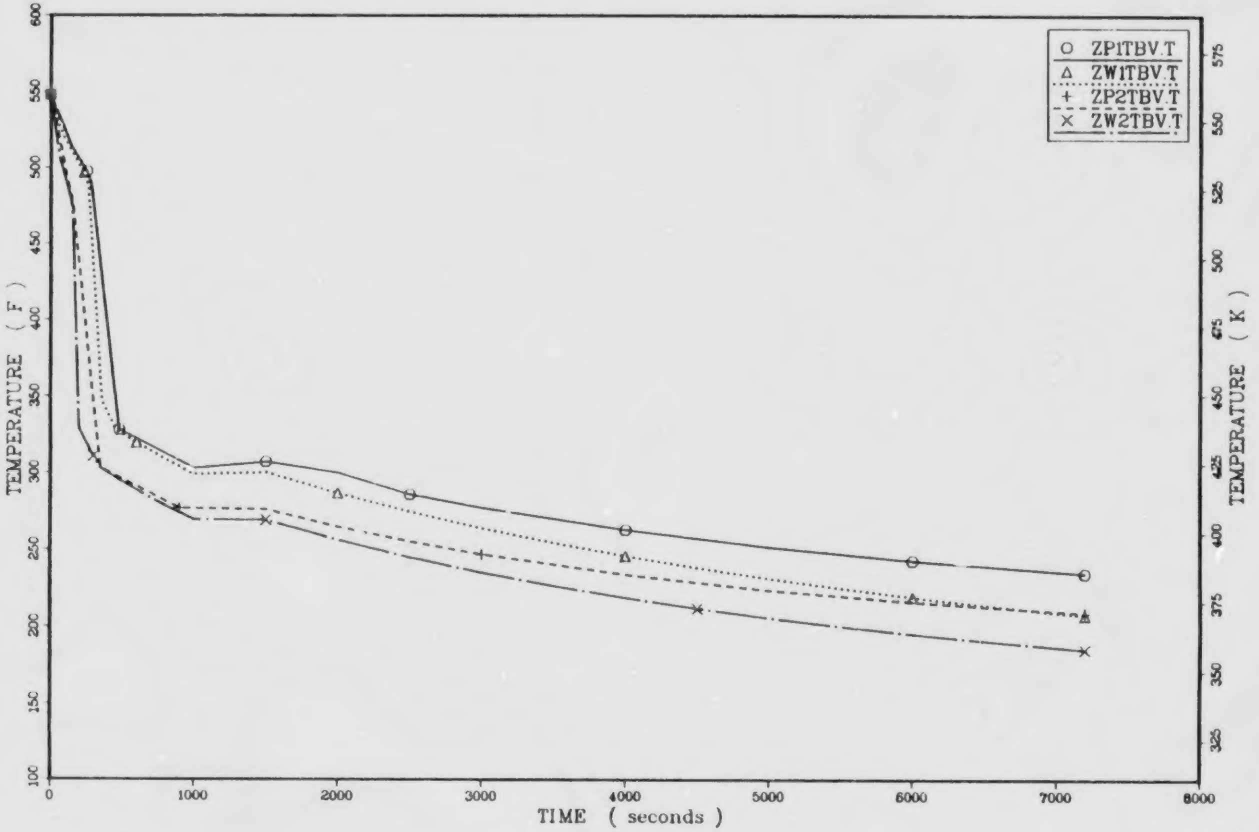

Figure 4.65 Extrapolated downcomer temperatures for Oconee-1 TBV failure at hot stand-by cases.

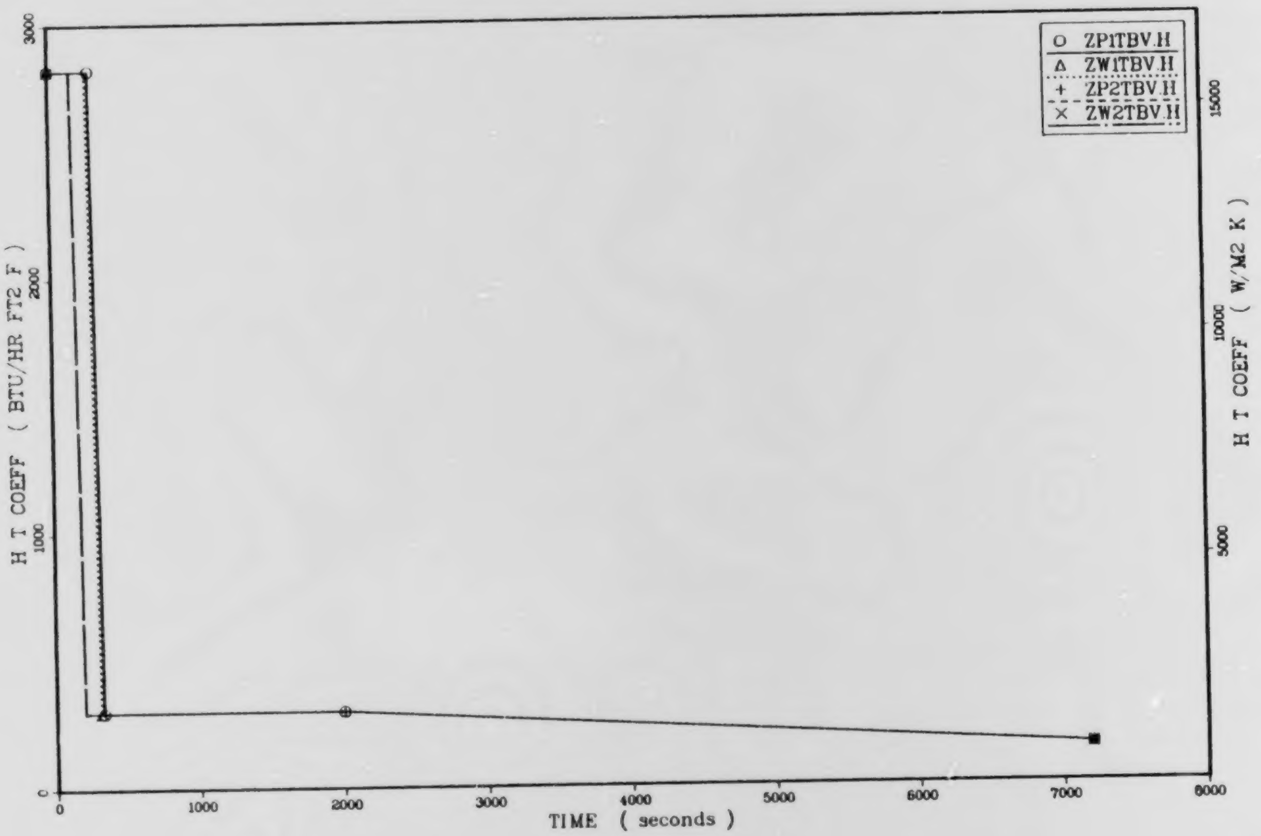

Figure 4.66 Extrapolated downcomer heat transfer coefficients for Oconee-1 TBV failure at hot stand-by cases.

Table 4.22 PORV-sized LOCA cases and groupings

	Case SBLOCA-								
Condition[*]	1	2	3	4	5	6	7	8	9
Secondary Pressure Controls									
Operates as designed	x	x	x	x	x		x	x	x
2 TBVs fail open loop A						x			
Feedwater									
MFW runback as designed	x					x	x		
SGA runback failure		x		x				x	
LOMFW at time = 0.0			x		x				x
EFW operates as designed	x	x	x			x	x	x	x
EFW in manual				x	x				
Primary Pressure Controls									
PORV block valve open	x	x	x	x	x	x	x	x	x
PORV operates as designed[**,***]	x	x	x	x	x	x	x	x	x
SRV fails to reseat[**]							x	x	x
Operator Actions									
RCP tripped at HPI + 30 s	x	x	x	x	x	x	x	x	x
Full EFW flow after SG dryout				x	x				
Isolates SGs at 20 min						x			
Group number (PSBG-)	1	2	3	2	4	5	1	2	3

[*] Other conditions common to all cases: full power, decay heat 1.0 times ANS, all ECCS components operate as designed

[**] If challenged

[***] Break is PORV-sized and may occur anywhere

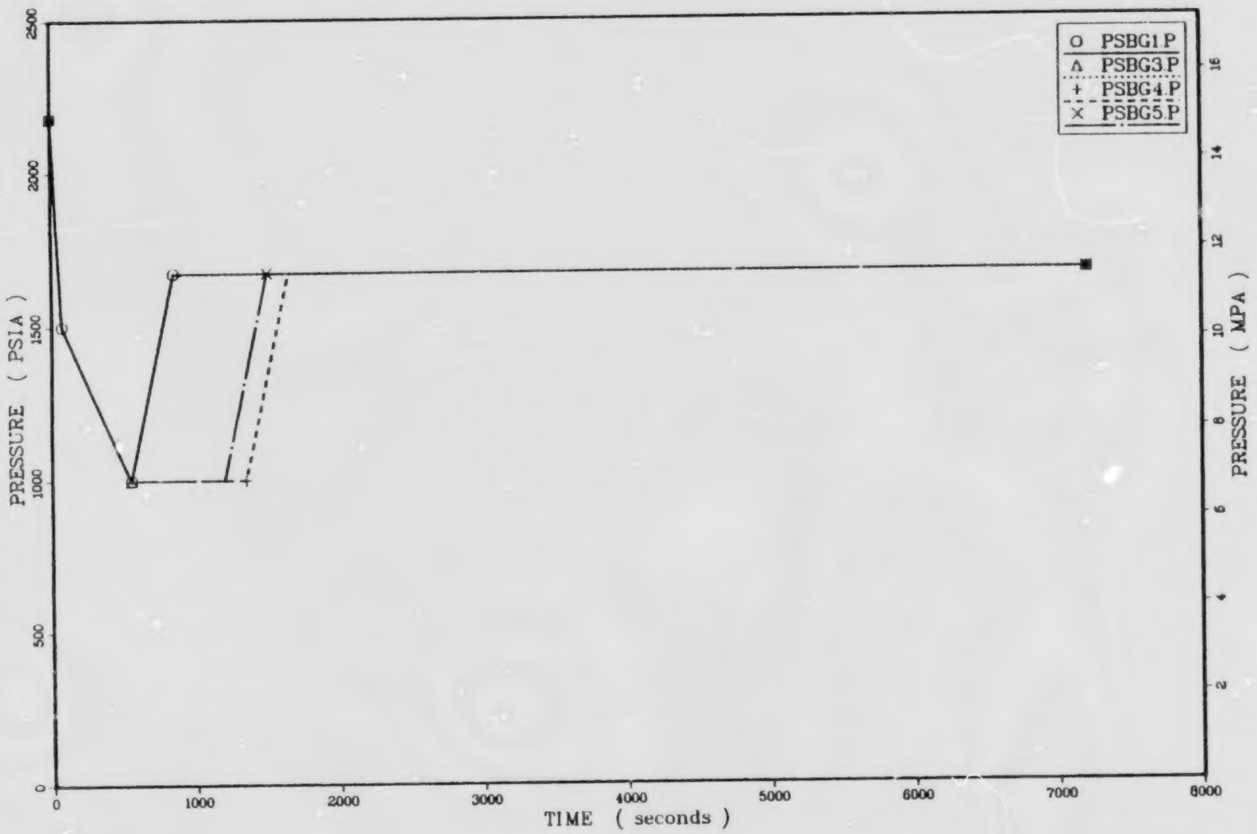

Figure 4.67 Extrapolated downcomer pressures for Oconee-1 PORV-sized LOCA cases.

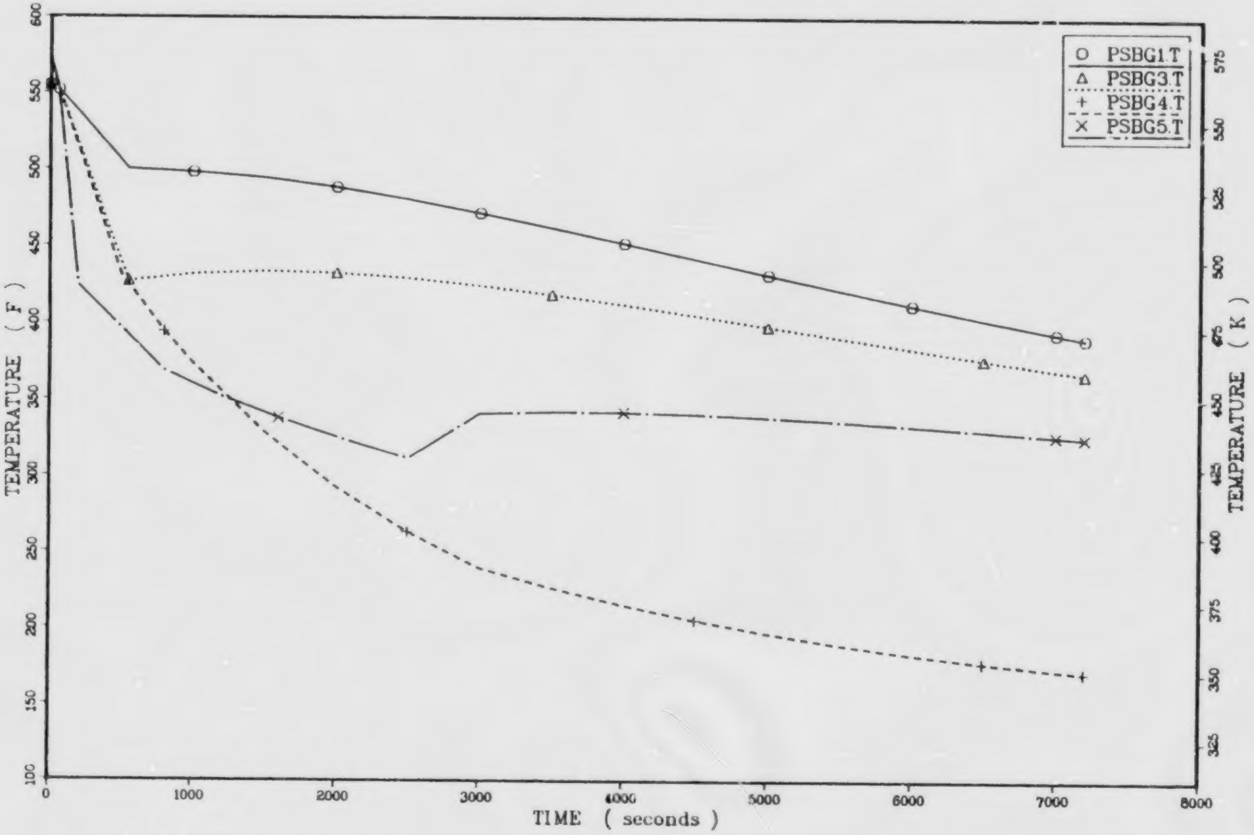

Figure 4.68 Extrapolated downcomer temperatures for Oconee-1 PORV-sized LOCA cases.

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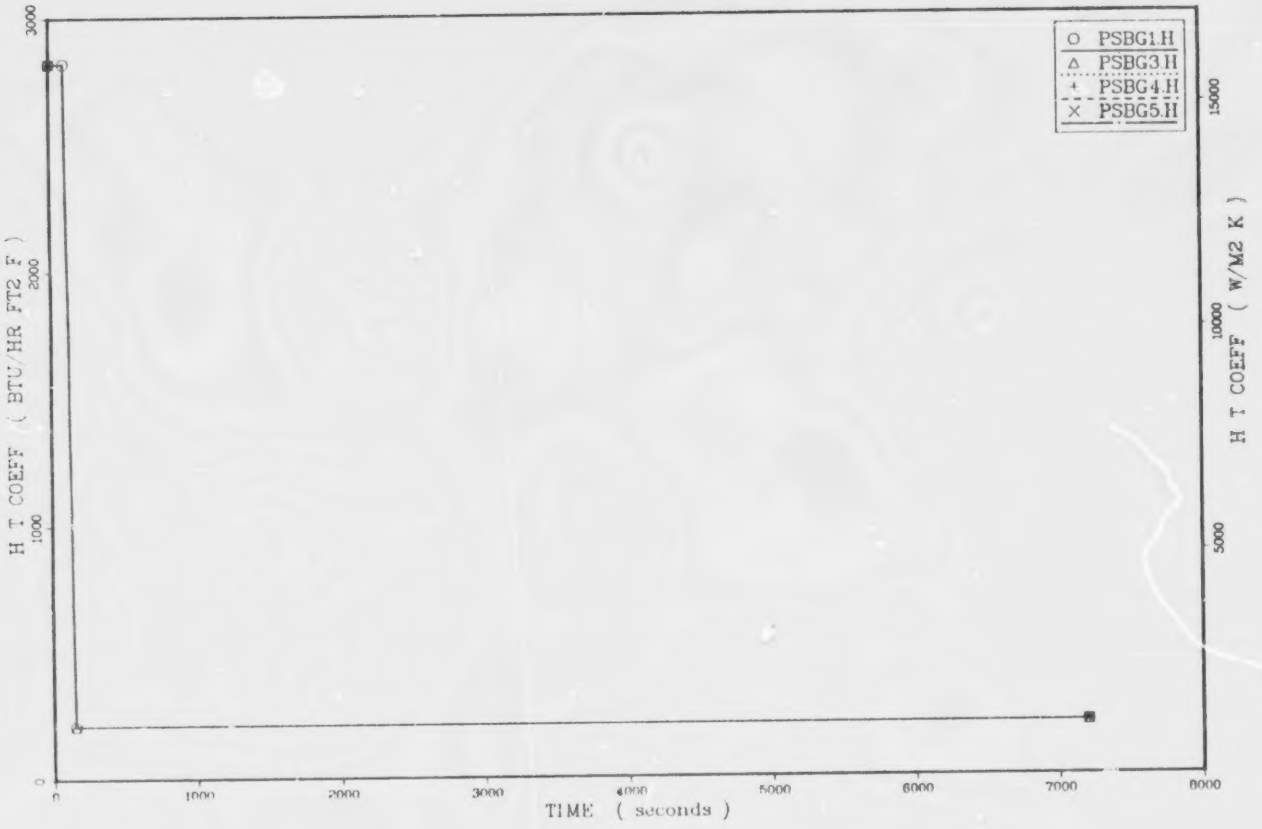

Figure 4.69 Extrapolated downcomer heat transfer coefficients for Oconee-1 PORV-sized LOCA cases.

feedwater operation (PSBG1) and MFW overfeed with trip on SG level (PSBG2) have the same influence since the steam generators are filled once and only once. Group 3 starts out cooler initially because of the exclusive use of cold EFW to fill the steam generators to the level for natural circulation. The steam generators are, in effect, totally isolated and do not contribute further to system cooldown. In group 4, the EFW level control fails and the steam generators flood out, allowing the steam generators to work with the HPI to cool the system to very low temperatures. In group 5, the isolation of the stuck-open TBVs allows a temperature recovery upon loss of this cooldown mechanism. The minimum temperatures for cases PSBG1 through PSBG5 are 472 K, 474 K, 459 K, 350 K, and 436 K. The final downcomer pressure in each case is 11.5 MPA.

4.5.5 Feedwater Overfeed Cases

Seven feedwater overfeed cases were originally identified for evaluation. Later, a very mild eighth case was added. The seven cases appear as transients 16 (FW1 or OVRFD1) through 22 (FW7 or OVRFD7) in Table 4.25. Table 4.23 presents the conditions for these cases. Figures 4.70 through 4.72 give the extrapolated pressures, temperatures, and heat transfer coefficients for cases 1, 3, 6, 7, and the new case number 8 (loss of main feedwater with subsequent refill with EFW controlled to level). With the exception of case 6, all of the transients are very mild. Cases 2 and 4 are similar to case 1. Case 6 achieves low temperatures because of an extensive EFW overfeed similar to the Rancho Seco-type transient. Minimum temperatures for cases FW1 through FW8 are 560 K, 560 K, 530 K, 550 K, 560 K, 419 K, 537 K, and 560 K.

Table 4.23 Feedwater transient cases

Condition*	Case FW- (or OVRFD-)						
	1	2	3	4	5	6	7
Main Feedwater							
SGA runback failure	x	x			x		
Both SG runback failure			x	x			
LOMFW at time = 0.0						x	x
MFWP trip high SG level	x	x	x	x			
MFWP high SG level trip failure					x		
Emergency Feedwater							
EFW works as designed	x	x	x	x	x		
EFW on manual						x	x
Operator Actions							
RCP trip at HPI + 30 s	x		x		x	x	x
Fail to trip RCP		x		x			
EFW restored full flow at 20 min						x	
MFW restored full flow at 20 min							x

*Conditions common to all cases: full power, decay heat 1.0 times ANS, secondary pressure control systems operate as designed, primary pressure control system operate as designed, all ECCS components operate as designed

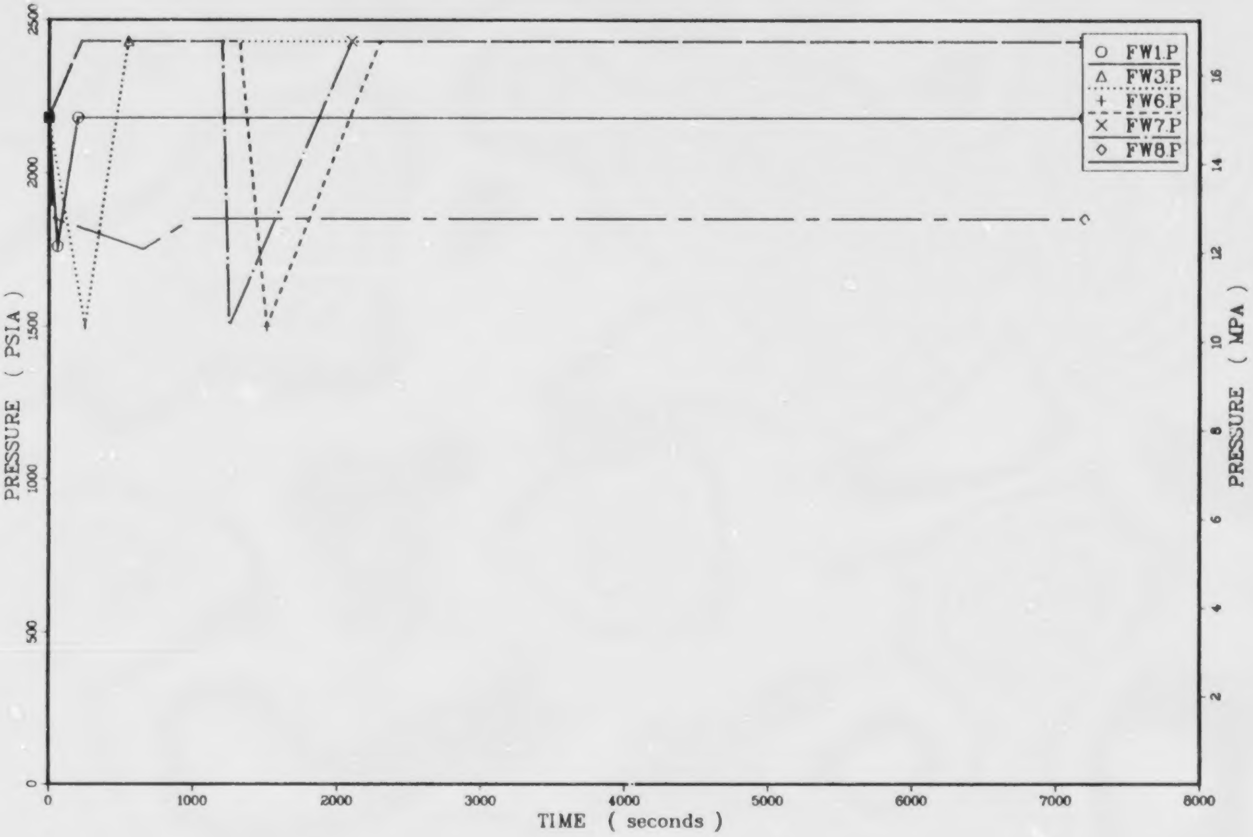

Figure 4.70 Extrapolated downcomer pressures for Oconee-1 feedwater transient cases.

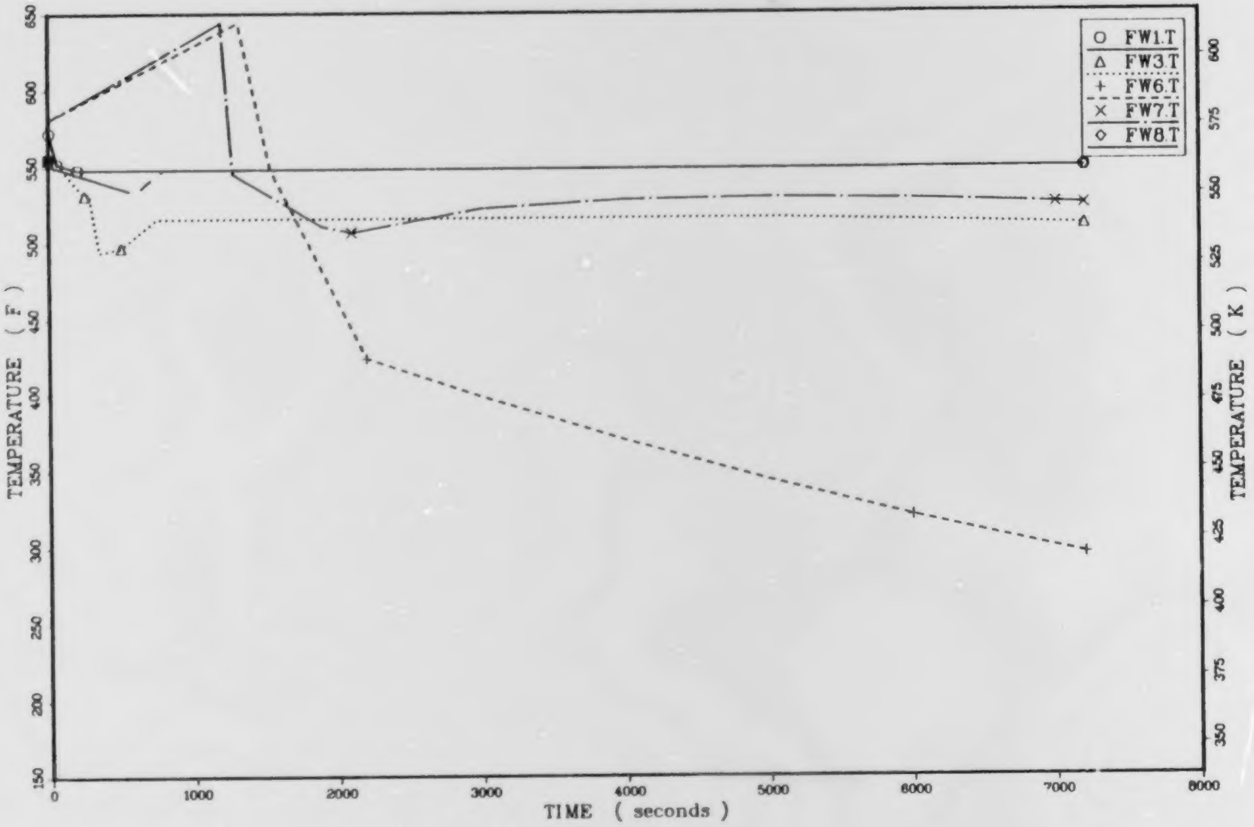

Figure 4.71 Extrapolated downcomer temperatures for Oconee-1 feedwater transient cases.

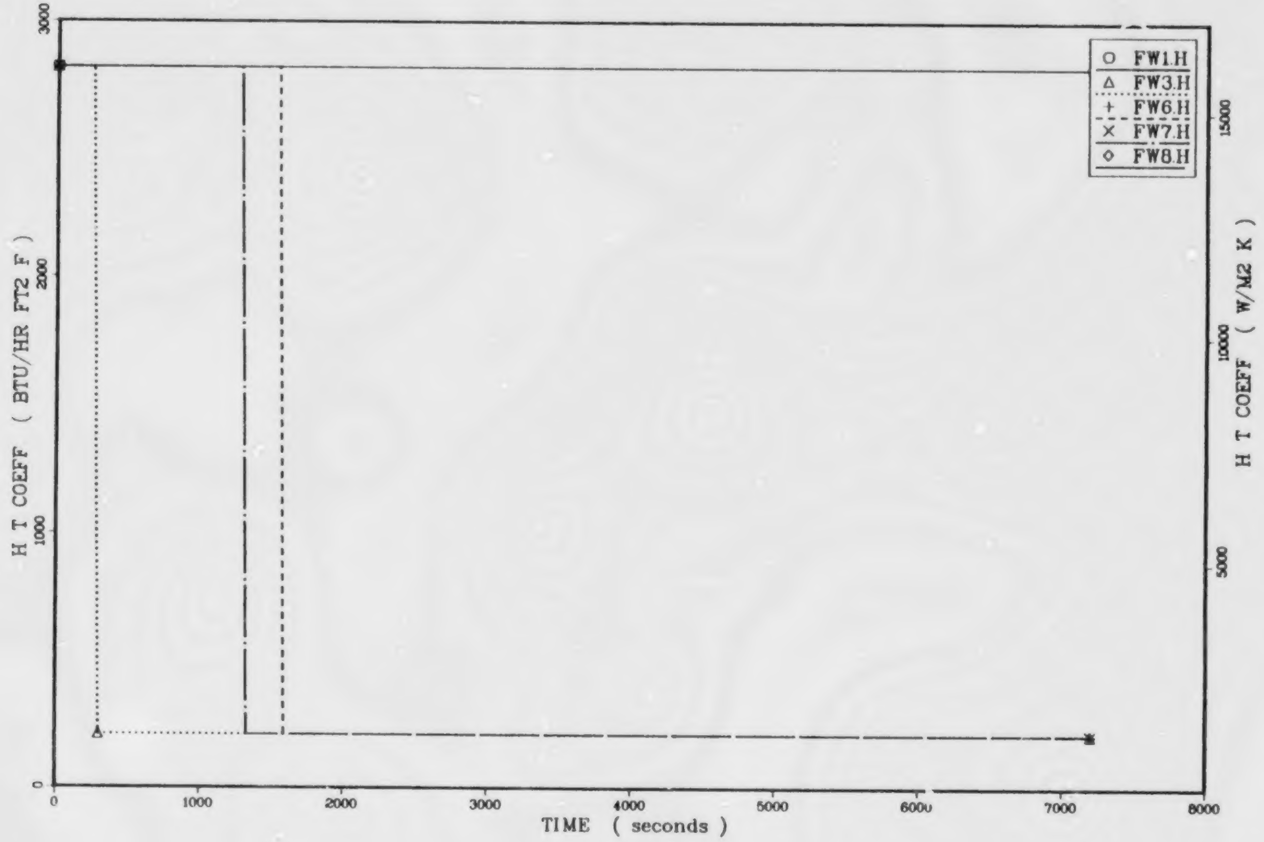

Figure 4.72 Extrapolated downcomer heat transfer coefficients for Oconee-1 feedwater transient cases.

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4.5.6 Steam Generator Tube Rupture

The steam generator tube rupture is essentially a small-break LOCA that dumps the lost primary coolant into the steam generator. For a single tube rupture, the INEL SGTR calculation in Section 4.3.1.9 applies directly. For multiple tube ruptures, the PORV-sized LOCA extrapolations would generally be representative. In either event, the temperature and pressure response of the Oconee-1 plant should be mild as far as PTS is concerned.

4.5.7 Main Steam Line Break Cases

Seven main steam line break cases were evaluated. Table 4.24 lists the conditions for these cases. The sequences comprise transients 26 (MSLB1) through 32 (MSLB7) respectively in Table 4.25. Figures 4.73 through 4.75 present the extrapolated pressures, temperatures, and heat transfer coefficients for cases MSLB1, MSLB3, MSLB4, and MSLB6. Cases MSLB2 and 7 correspond closely to MSLB1. Likewise, MSLB5 is very similar to MSLB3. Extrapolation of these sequences was problematic because of the disparity of specifications between the RELAP5 and TRAC detailed calculations and these extrapolated cases.

Most notably, the 20-min delay of feedwater isolation, RCP restart, and HPI for the extrapolated cases represented a very long delay relative to the early RCP restarts and isolation of feedwater at 10 min in the LANL and INEL main steam line break calculations. The rapid cooldown rates in the detailed calculations (see Sections 4.3.1.4 and 4.3.2.1) could not possibly be sustained for 20 min. The cooldown model in Appendix C could not be applied because the complex two-phase heat transfer regime in the affected

Table 4.24 Main steam line break study cases

Condition*	Case MSLB- 1	2	3	4	5	6	7
Secondary Pressure Control							
TSVs close on turbine trip signal	X	X	X	X	X	X	X
TBVs operate as designed	X	X	X	X	X	X	X
SSRVs operate as designed (if challenged)	X	X	X	X	X	X	X
Feedwater							
MFW runback controlled by ICS	X			X			X
MFW overfeed both steam generators			X				
MFW overfeed to broken steam generator		X					
MFW pump trip on high steam generator level (HW pumps, CB pumps continue to operate)		X	X				
MFW pumps trip at time = 0.00					X	X	
EFW initiates on set point	X	X	X	X	X		X
EFW fails to initiate automatically						X	
EFW controlled to level	X	X	X		X		X
EFW overfeed to broken steam generator				X			
Primary Pressure Control							
PORV block valve open	X	X	X	X	X	X	X
PORV operates as designed**	X	X	X	X	X	X	
PORV fails to reseat**							X
SRVs operated as designed**	X	X	X	X	X	X	X

Table 4.24 (continued)

Condition*	Case MSLB-						
	1	2	3	4	5	6	7
Operator Actions							
RCPs tripped at HPI + 30 s	X	X	X	X	X	X	X
Both SGs isolated at 20 min	X	X	X	X	X		X
Both SG's isolated at 5 min						X	
EFW and TBVs restored to intact SG at 21 min	X	X	X	X	X		X
EFW and TBVs restored to intact SG at 6 min						X	
HPI throttled at later of 22 min or 50°F subcooling	X	X	X	X	X	X	X
1 RCP/loop restarted at later of 22 min or 50°F subcooling	X	X	X	X	X	X	X
Failed PORV isolated when system pressure drops below 2400 psi							X

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*Other conditions common to all cases: full power, decay heat 1.0 times ANS, ECCS systems operate as designed

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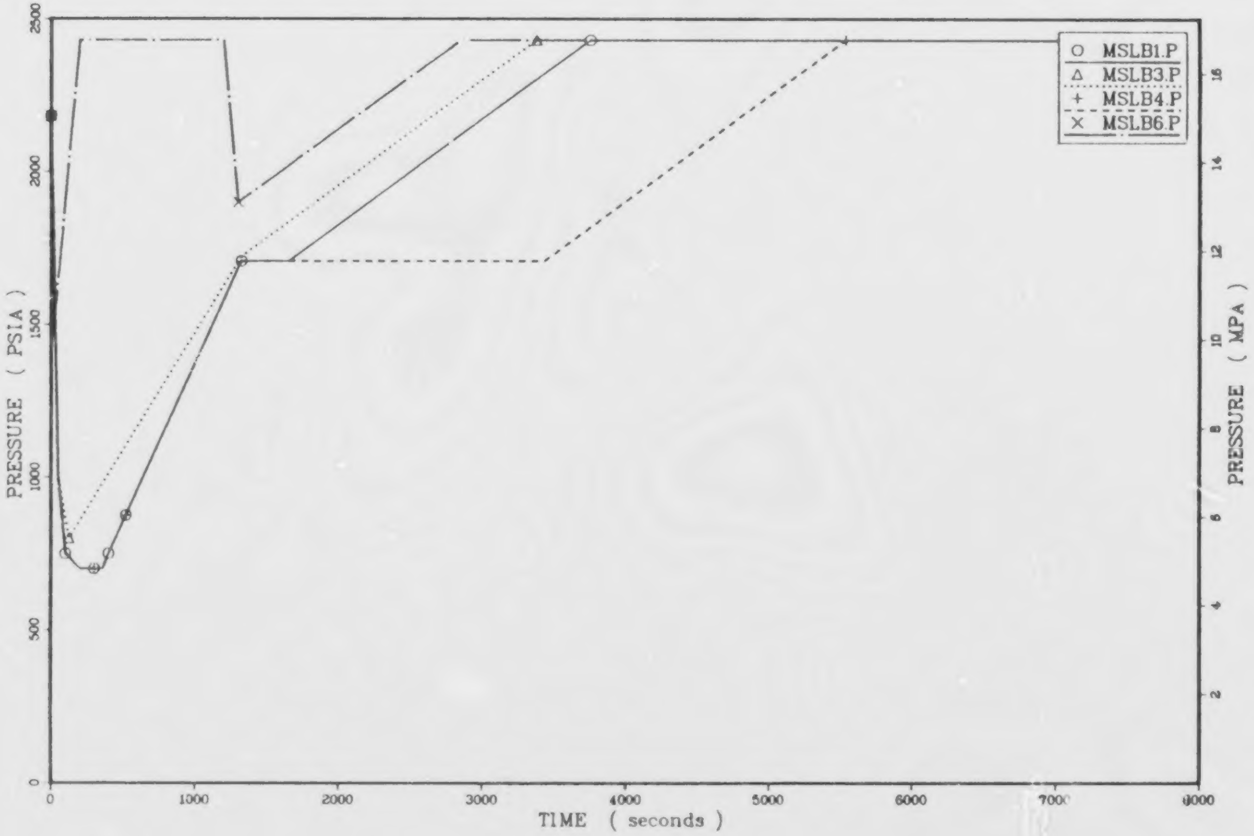

Figure 4.73 Extrapolated downcomer pressures for Oconee-1 main steam line break cases.

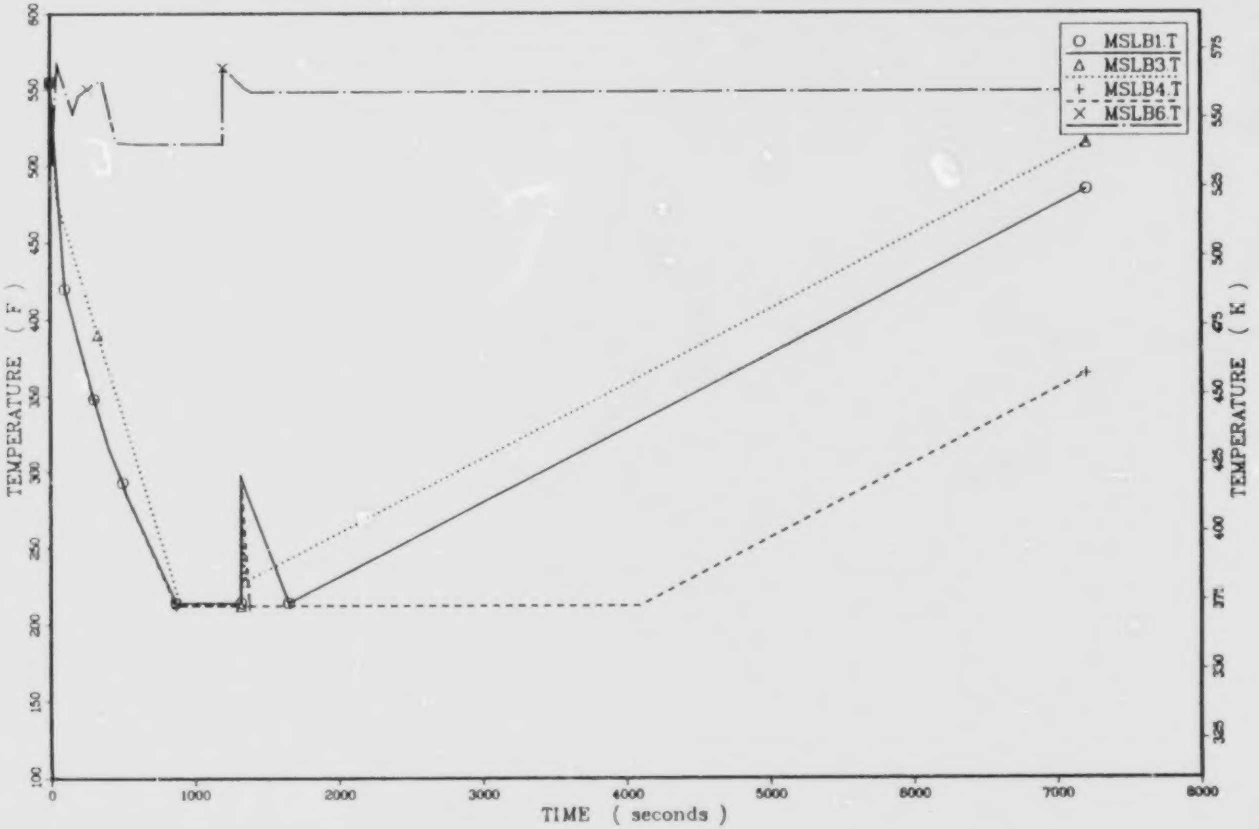

Figure 4.74 Extrapolated downcomer temperatures for Oconee-1 main steam line break cases.

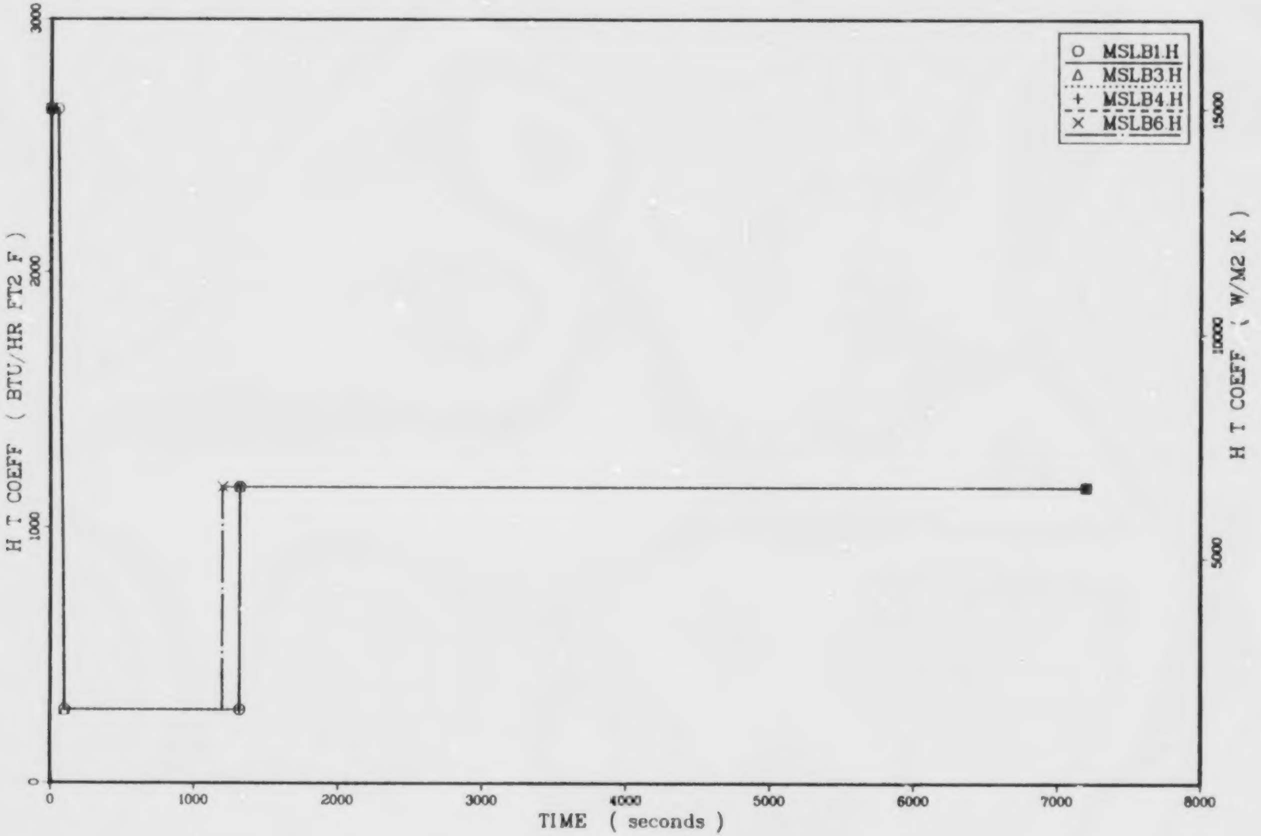

Figure 4.75 Extrapolated downcomer heat transfer coefficients for Oconee-1 main steam line break cases.

steam generator violates the model's simplifying assumptions (see Appendix C, Section C.2.3). Also, the natural circulation flow data necessary to set up the model were cut short by the early restart of the reactor cooling pumps.

As a result, a very simple approach was taken to extrapolate these cases. First, the LANL main steam line break cases were selected as the bases for extrapolation, as they predicted the most severe cooldown rates. Next, the downcomer temperature response of the extrapolated cases was taken to be similar to the portion of the LANL case prior to the restart of the RCPs at 526 s. Continuing on the same slope, the downcomer temperature was extrapolated to a minimum temperature of 373 K (212°F). This minimum temperature is considered a conservative physical limit.

The temperature remains at this minimum limit until the steam generator dries out. The entrainment of liquids from the broken steam generator is assumed to abate at low temperatures, allowing the steam generator to refill. The time to steam generator dryout reflects the time needed to accumulate sufficient decay heat and RCP power dissipation to complete the steam generator dryout. Reheating and repressurization trends from the LANL cases were employed to complete the extrapolation after steam generator dryout.

The temperature extrapolations for MSLB1, MSLB3, and MSLB4 in Figure 4.74 were based on the above assumptions. The differences in these curves can be ascribed to the quantity of water accumulated in the broken steam generators prior to isolation. The MSLB6 case does not experience overcooling due to the loss of all feedwater followed by early isolation of the broken steam generator and controlled cooldown using the intact steam generator. See Appendix C, Section C.3, for detailed discussion of these transients.

4.6 Summary of Thermal-Hydraulic Evaluations

Pressure, temperature, and heat transfer coefficient response profiles were calculated and assigned for every event tree end-state having a frequency of occurrence greater than 10^{-6} per year. The use of sequence similarity groupings further reduced the number of sequences evaluated.

Detailed calculations using the RELAP5 and TRAC models provided the data upon which sequence extrapolations were developed. The detailed calculations included event initiators known or suspected to result in significant overcooling responses. These initiators included steam line breaks, turbine bypass valve failures, feedwater overfeeds, steam generator tube ruptures, and small primary breaks. The RELAP5 and TRAC models developed for this study were, at the time, the most complex applications of the codes to date.

Extrapolations were based on the results of the detailed calculations. Where specified conditions of an extrapolated transient closely matched those of a calculated sequence, the calculated results (i.e., pressure and temperature trends) were used over the applicable portion of the extrapolation. Temperature extrapolations were facilitated by the use of a cooldown model, which accounted for all major heating and cooling mechanisms and the system mass and loop mass flows determining the time constants on the response.

The results of the extrapolations and detailed calculations are summarized on Table 4.25. The transient numbers and identifications correlate to the nomenclature used in the fracture mechanics (Chapter 5.0) and results (Chapter 6.0) sections. The minimum downcomer temperatures and extent of primary repressurization facilitate references to the fracture mechanics calculation.

Table 4.25 Summary of Oconee-1 PTS thermal-hdraulic analyses

Transient Number	Transient Name	Description/ Alternate Name	Section Reference	Minimum Temperature T min (K)	Maximum Pressure at or after T min (MPA)
1	TBVG1		C.4.1	420	16.8
2	TBVG2		C.4.2	471	16.8
3	TBVG3		C.4.3	462	16.8
4	TBVG4		C.4.4	365	16.8
5	TBVG5		C.4.5	402	16.8
6	TBVG6		C.4.6	398	11.5
7	TBVG7		C.4.7	406	16.8
8	TBVG8		C.4.8	449	16.8
9	TBVG9		C.4.9	365	16.8
10	TBVG10		C.4.10	386	11.5
11	PSBG1		C.6.1	472	11.5
12	PSBG2		C.6.2	473	11.5
13	PSBG3		C.6.3	459	11.5
14	PSBG4		C.6.4	350	11.5
15	PSBG5		C.6.5	436	11.5
16	OVRFD1	FW1	C.7.1	562	15.0
17	OVRFD2	FW2	C.7.1	562	15.0
18	OVRFD3	FW3	C.7.2	530	16.8
19	OVRFD4	FW4	C.7.3	550	16.8
20	OVRFD5	FW5	C.7.4	562	15.0
21	OVRFD6	FW6	C.7.5	419	16.8
22	OVRFD7	FW7	C.7.6	536	16.8
23	SGTR		C.8.0	505	17.0
24	ZP1TBV		C.5.0	385	16.8
25	ZP2TBV		C.5.0	371	16.5
26	MSLB1		C.3.1	373	16.8

Table 4.25 (Continued)

Transient Number	Transient Name	Description/ Alternate Name	Section Reference	Minimum Temperature T min (K)	Maximum Pressure at or after T min (MPA)
27	MSLB2		C.3.2	373	16.8
28	MSLB3		C.3.3	373	16.8
29	MSLB4		C.3.4	373	16.8
30	MSLB5		C.3.5	373	16.8
31	MSLB6		C.3.6	541	16.8
32	MSLB7		C.3.7	373	16.8
33	W1TBV	ZW1TBV	C.5.0	370	16.8
34	W2TBV	ZW2TBV	C.5.0	358	16.8
35	LANL1	MSLB (base)	4.3.2.1	405	17.0
36	LANL2	(void)			
37	LANL3	MSLB (Para 2)	4.3.2.1	475	7.7
38	LANL4	MSLB (Para 3)	4.3.2.1	450	12.8
39	LANL5	PORV LOCA	4.3.2.2	528	11.5
40	LANL6	Oconee-3			
41	LANL7	TBV (5A)	4.3.2.3	365	17.0
42	LANL8	TBV (5B)	4.3.2.3	440	17.0
43	LANL9	TBV (5C)	4.3.2.2	430	4.0
44	LANL10	TBV (6A)	4.3.2.4	350	17.0
45	LANL11	TBV (6B)	4.3.2.4	465	17.0
46	LANL12	TBV (6C)	4.3.2.4	350	4.0
47	LANL13	LOCA 2"	4.3.2.5	425	2.8
48	LANL14	LOCA 4"	4.3.2.6	320	0.3
49	LANL15	Rancho Seco	4.3.2.7	452	17.0
50	LANL16	(void)			
51	INEL1	MSLB (Revised)	4.3.1.4	494	17.0
52	INEL2	PORV LOCA	4.3.1.3	545	11.4

Table 4.25 (Continued)

Transient Number	Transient Name	Description/ Alternate Name	Section Reference	Minimum Temperature T min (K)	Maximum Pressure at or after T min (MPA)
53	INEL3	Overfeed	4.3.1.2	505	17.0
54	INEL4	Oconee-3			
55	INEL5	LOCA 2"	4.3.1.7	355	1.5
56	INEL6	Max. Sus. MFW	4.3.1.5	500	17.0
57	INEL7	4TBV-HS	4.3.1.6	387	17.0
58	INEL8	Pump Suction 2.5"	4.3.1.8	446	5.2
59	INEL9	SGTR	4.3.1.9	505	17.0

REFERENCES

4.1 C. D. Fletcher and others, "RELAP5 Thermal-Hydraulic Analysis of Pressurized Thermal Shock Sequences for the Oconee-1 Pressurized Water Reactor," EGG-NSMD-63343, July 1983.

4.2 J. Ireland and others, "TRAC Analysis of Severe Overcooling Transients for the Oconee-1 PWR," LA-UR-83-3182, (no date).

4.3 T. G. Theophanous, Purdue University, letter to J. D. White, ORNL, December 22, 1983.

5.0 CONDITIONAL PROBABILITY OF VESSEL FAILURE

R. D. Cheverton, Oak Ridge National Laboratory

5.1 Introduction

The conditional probability of vessel failure (through-wall cracking) was calculated for the Oconee-1 reactor pressure vessel. The necessary models for performing a probabilistic fracture-mechanics analysis for the Oconee-1 reactor pressure vessel and the results of the analysis are discussed.

5.2 Description of Basic Problem

During an overcooling transient in a pressurized-water reactor (PWR), the reactor pressure vessel is subjected to thermal shock in the sense that thermal stresses are created in the vessel wall as a result of rapid removal of heat from its inner surface. The thermal stresses are superimposed on the pressure stresses with the result that the net stresses are positive (tensile) at and near the inner surface of the wall and are substantially lower and perhaps negative elsewhere, depending on the magnitude of the pressure stress. The concern over the high tensile stresses near the inner surface is that they result in high stress intensity factors (K_I) for inner-surface flaws that may be present. To compound the matter, the reduced temperature associated with the thermal shock and radiation damage result in relatively low fracture toughness values for the vessel material, particularly near the inner surface. Thus, there is a possibility of propagation of initially very shallow as well as deeper flaws, and the probability increases with time because of the time dependence of radiation damage.

The positive gradient in temperature and the negative gradients in stress and fluence through the wall tend to provide a mechanism for crack arrest. Even so, if the crack is very long on the surface and propagates deep enough, the remaining vessel ligament will become plastic, and the vessel internal pressure will ultimately result in rupture of the vessel. Thus, for each thermal transient there will be a maximum permissible pressure that is a function of the time that the vessel has been in operation.

Crack propagation may also be limited by a phenomenon referred to as warm prestressing (WPS), which has been demonstrated in the laboratory with small specimens[1] and also in a rather large, thick-walled cylinder during a thermal-shock experiment.[2] In such cases, WPS simply refers to the inability of a crack to initiate while K_I is decreasing with time, that is, while the crack is closing. While this special situation is encountered during some specific overcooling accidents, caution must be exercised in taking credit for WPS because changes in the pressure that affect little else can delay or eliminate the requisite conditions for WPS.

The area of the vessel of particular concern in the event of an overcooling accident (OCA) is the so-called beltline region, that is, the

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area directly across from the core where (1) the radiation damage is the greatest, (2) the thermal shock could be severe, and (3) a rupture of the vessel could preclude flooding of the core. Whether or not a particular degree of rupture associated with a particular transient could in fact preclude flooding of the core has not been determined. For the purpose of this report, it is sufficient to predict whether a flaw will propagate completely through the wall of the vessel.

The radiation-induced reduction in fracture toughness of the vessel material is primarily a function of the fast-neutron fluence and the concentrations of copper (a contaminant) and nickel (an alloying element). In most PWR vessels the highest concentrations of copper are found in the welds that join the segments of the vessel, and many of these welds have high concentrations of nickel as well. Thus, for some PWR vessels the welds are of primary concern, and the Oconee-1 reactor pressure vessel falls in this category.

The beltline region of reactor pressure vessels is fabricated using either forged-ring segments or rolled-plate segments. Vessels made with forgings have only circumferential welds, while plate-type vessels have both circumferential and axial welds, as shown in Figure 5.1. For plate-type vessels with staggered axial welds and for which radiation damage is much more severe in the welds than in the base material, the final surface length of a propagating inner-surface axial flaw tends to be limited to the length of the axial weld in which it resides, that is,

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the height of the shell course. Furthermore, only that portion of a weld that is within the axial bounds of the core need be considered because of the steep attenuation of the fast-neutron flux beyond the fuel region.

The extended length of an initially short, circumferentially oriented flaw located in a circumferential weld also tends to be limited because of an azimuthal variation in the fast-neutron flux (see Figure 5.1).

The behavior of an assumed flaw can be predicted for a given transient using fracture-mechanics methods of analysis. In such an analysis the parameters and considerations involved are the size, shape, and orientation of the flaw; the thermal and pressure stresses resulting from a specific transient; the temperature and fast-neutron fluence distributions throughout the vessel wall; the effect of fluence and material chemistry on radiation damage; a variety of material properties; and a comparison of the stress intensity factor (K_I) associated with the tip of the flaw with the material's static crack-initiation and crack-arrest fracture-toughness values (K_{Ic} and K_{Ia}). Each of these factors must be considered in the development of an appropriate analytical model for evaluating the integrity of a PWR vessel when subjected to OCA loading conditions. The necessary models for performing a probabilistic fracture-mechanics analysis for the Oconee-1 reactor pressure vessel and the results of the analysis are discussed in the remainder of this chapter.

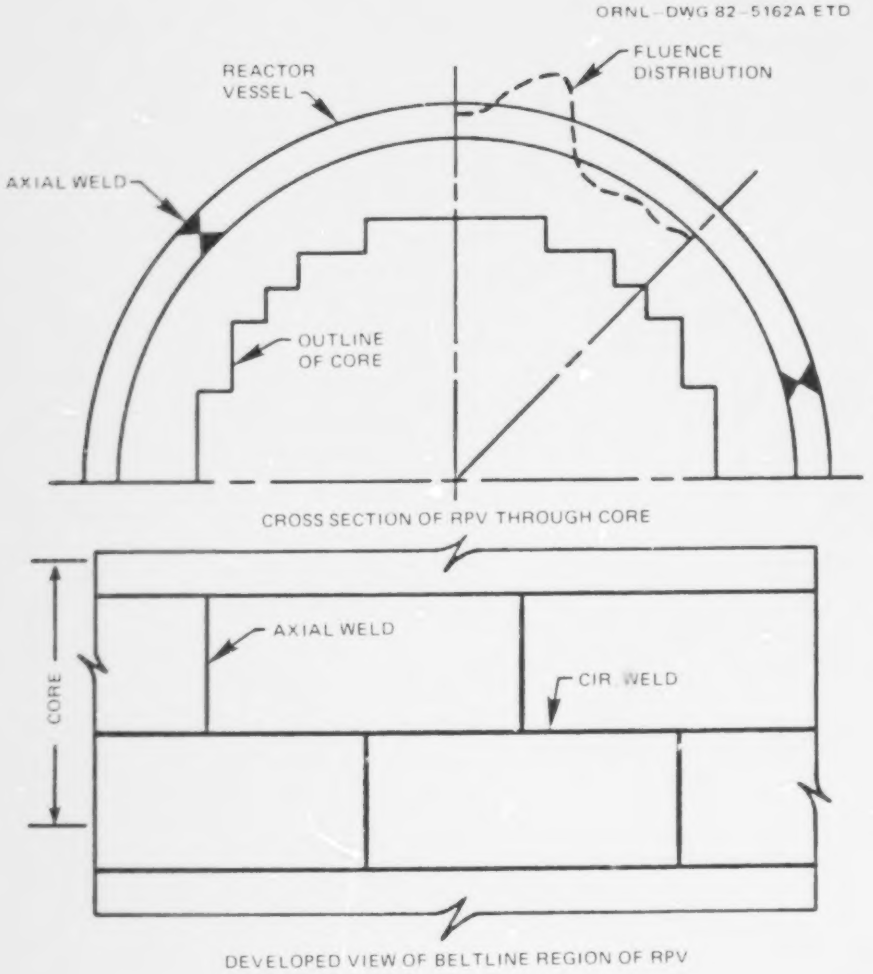
ORNL-DWG 82-5162A ETD
REACTOR VESSEL
FLUENCE DISTRIBUTION
AXIAL WELD
OUTLINE OF CORE
CROSS SECTION OF RPV THROUGH CORE
AXIAL WELD
CIR. WELD
CORE
DEVELOPED VIEW OF BELTLINE REGION OF RPV

Figure 5.1. Cross section and developed view of plate-type PWR pressure vessel.

5.3 Calculational Models

The conditional probability of vessel failure (through-wall cracking) was calculated for the Oconee-1 reactor pressure vessel using the OCA-P code.[3] OCA-P accepts as input the primary system pressure, the temperature of the coolant in the reactor vessel downcomer, and the fluid-film heat transfer coefficient adjacent to the vessel wall, all as a function of time in a specified OCA transient. The code then performs one-dimensional thermal and stress analyses for the vessel wall and finally a probabilistic fracture-mechanics analysis. Details of OCA-P necessary for an understanding of the Oconee-1 vessel analysis are discussed below.

5.3.1 Fracture-Mechanics Model

The fracture-mechanics (FM) model in OCA-P is based on linear elastic fracture mechanics (LEFM) and uses a specified maximum value of K_{Ia} to account for upper-shelf behavior. The stress intensity factor (K_I) is calculated using superposition techniques in conjunction with influence coefficients that were calculated using finite-element techniques. The application of this procedure makes it possible to perform a large number of deterministic FM calculations at reasonable cost, a necessary condition for performing the probabilistic analysis.

The Oconee-1 vessel was fabricated from sections of plate and has both axial and circumferential welds in the beltline region, as shown in Figure 5.2. Several of the welds have rather high concentrations of copper and nickel while the concentration of copper in the base material is

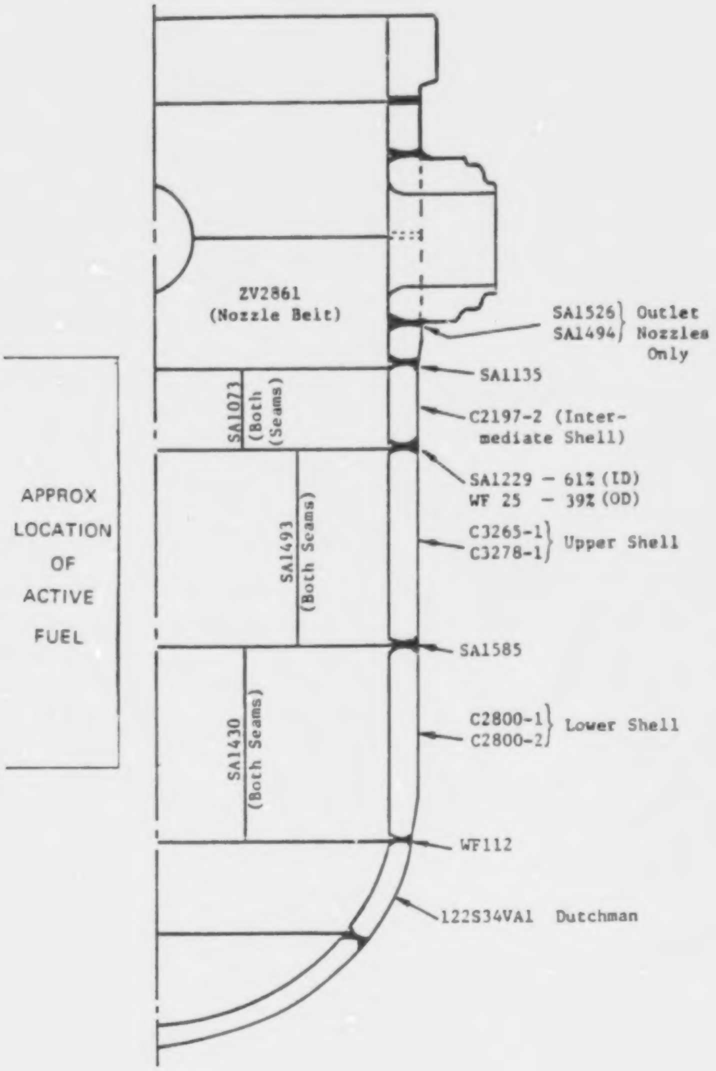
ZV2861
(Nozzle Belt)
SA1526
SA1494
Outlet
Nozzles
Only
SA1135
SA1073
(Both
Seams)
C2197-2 (Inter-
mediate Shell)
SA1229 – 61% (ID)
WF 25 – 39% (OD)
APPROX
LOCATION
OF
ACTIVE
FUEL
SA1493
(Both Seams)
C3265-1
C3278-1
Upper Shell
SA1585
SA1430
(Both Seams)
C2800-1
C2800-2
Lower Shell
WF112
122S34VA1 Dutchman

Figure 5.2. Location and identification of materials used in in fabrication of Oconee-1 reactor pressure vessel.

significantly lower. Furthermore, values of K_I for deep axial flaws are much greater than for deep circumferential flaws. Thus as discussed in Section 5.4 and Appendix D in greater detail, the flaws of interest are restricted to the axially oriented welds.

The length of flaws in the axial welds with depths greater than ~40 mm was assumed to be 1.8 m, which is approximately the height of a shell course, and the shape was assumed to be semielliptical (this flaw is referred to as the 2-m flaw). Since the ends of this flaw are fixed, propagation was judged on the basis of the K ratios (K_I/K_{Ic}, K_I/K_{Ia}) at the deepest point of the flaw.

Shallower flaws were assumed to be two-dimensional (termed 2-D flaws), since long shallow flaws are essentially two-dimensional, and short flaws tend to grow on the surface to become long flaws,[4] at least in the absence of cladding. Because the effect of cladding on the surface extension of short flaws is not known at this time, any possible beneficial effect has been discounted.

Cladding on the inner surface of PWR pressure vessels was included in the OCA-P analysis as a discrete region to the extent that the thermal and stress effects were accounted for. As mentioned above, the effect of cladding on the surface extension of finite-length flaws was not considered.

Because of the difference in the coefficient of thermal expansion between the cladding and base material, the calculated stresses in the cladding exceed the yield strength of the cladding by an appreciable amount, and this results in an overestimation of the K_I values for the flaws, which were assumed to terminate in the cladding or extend through the cladding into the base material. An alternative approach would be to limit the stress in the cladding to the yield stress, but this underestimates K_I because K_I is sensitive to the strain, which is not limited by the yielding phenomenon. The difference in K_I between these two extremes is not large; thus the conservative extreme was selected.

Material properties required for the fracture-mechanics analysis include the static crack initiation and arrest toughness values (K_{Ic} and K_{Ia}) and the nil-ductility reference temperature (RTNDT). For the probabilistic fracture-mechanics analysis, mean values of these parameters are required, and they were obtained for the vessel weld material as follows:

$$\overline{K}_{Ic} = 1.43\ \{36.5 + 3.084 \exp [0.036\ (T - \text{RTNDT} + 56)]\},\ \text{MPa}\sqrt{\text{m}} \tag{5.1}$$

$$\overline{K}_{Ia} = 1.25\ \{29.5 + 1.344 \exp [0.0261\ (T - \text{RTNDT} + 89)]\},\ \text{MPa}\sqrt{\text{m}} \tag{5.2}$$

where the quantity in braces represents the ASME Section XI[5] lower-bound toughness value and T is the temperature at the tip of the flaw in °C. These expressions were obtained by letting the ASME lower bound curves represent the mean values minus two standard deviations (2σ) and by letting $\sigma_{(K_{Ic})} = 0.15\ \overline{K}_{Ic}$ and $\sigma_{(K_{Ia})} = 0.10\ K_{Ia}$.

In many cases, if crack arrest takes place, it must do so at upper-shelf temperatures, that is, at temperatures that, under static loading conditions, result in ductile rather than brittle behavior of the material. Crack arrest under these conditions is not well understood but has been included in an approximate manner by specifying a maximum value of K_{Ia} that corresponds to the upper portion of an upper-shelf tearing resistance curve. As illustrated in Figure 5.3, which is a plot of K vs crack depth (a) and temperature (T) at a specific time in a transient, if the load line (K_I vs a, T) intersects the K_{Ia} curve at $K_{Ia} < (K_{Ia})_{max}$, upper-shelf temperatures are not encountered. If, on the other hand, the load line misses the rising portion of the K_{Ia} curve and then decreases, as it does for some transients, there is, according to the model, a possibility of crack arrest at upper-shelf temperatures.

The tearing resistance curve selected for this study represents a specific high copper, low-upper-shelf weld material that had been irradiated to a fluence of $\sim 1.2 \times 10^{19}$ neutrons/cm^2 at a temperature of $\sim 300°C$ and tested at 200°C.[6] The upper, nearly flat portion of this curve corresponds to a K_J value of ~ 220 MPa $\sqrt{m}$, and this value was used for $(K_{Ia})_{max}$; K_J was obtained using the relation

$$K_J = \sqrt{JE} \tag{5.3}$$

where J = strain energy release rate

E = Young's modulus

The tearing resistance of PWR vessel materials tends to decrease with increasing temperature and fluence, and thus the effects of temperature and

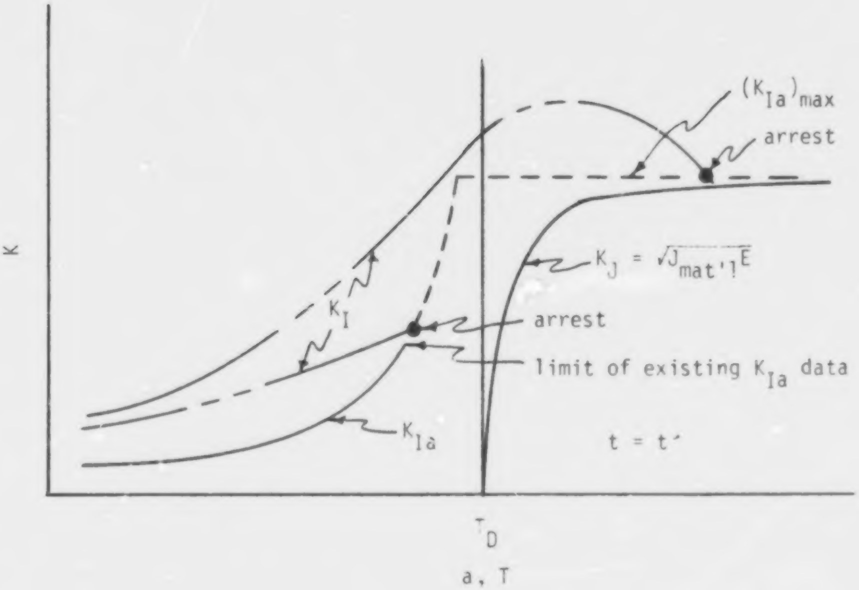

Figure 5.3. Illustration of a method of selecting $(K_{Ia})_{max}$.

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fluence tend to compensate for each other through the wall of the vessel. Because of this and the very approximate nature of the treatment of arrest on the upper shelf, no attempt was made to account more accurately for the effects of temperature and fluence on $(K_{Ia})_{max}$.

The nil-ductility reference temperature (RTNDT) is equal to the sum of an initial (zero fluence) value ($RTNDT_o$) and an increase due to radiation damage ($\Delta RTNDT$); that is,

$$RTNDT = RTNDT_o + \Delta RTNDT \tag{5.4}$$

The correlation for $\Delta\overline{RTNDT}$ used in these studies was recently proposed by Randall[7] and is

$$\Delta\overline{RTNDT} = 0.56\ [-10 + 470\ Cu + 350\ Cu\ Ni]\ (F \times 10^{-19})^{0.27}\ ,\ ^\circ C \tag{5.5}$$

or

$$\Delta\overline{RTNDT} = 0.56\ [283\ (F \times 10^{-19})^{0.194} - 48]\ ,\ ^\circ C \tag{5.6}$$

whichever is smaller, where

Cu, Ni = concentrations of copper and nickel, wt %,

F = fast-neutron fluence (neutron energy > 1 MeV), neutrons/cm^2

The attenuation of the fluence through the wall of the vessel is approximated with

$$F = F_o\ e^{-0.0094\ a} \tag{5.7}$$

where F_o is the fluence at the inner surface of the vessel and a is the crack depth in millimeters. The specific value of the coefficient in the exponent accounts to some extent for the effect of space-wise spectral changes on radiation damage.[7]

If the assumption is made that a short and shallow surface flaw can extend on the surface through the cladding to become a long flaw (and this assumption is made for these studies), then it must be assumed that under the proper circumstances a very shallow flaw that initially resides entirely within the cladding can propagate radially. Unfortunately, the fracture-toughness properties of the cladding material are very uncertain, and the few experimental data that are available indicate a radiation-induced reduction in fracture toughness similar to that for the base material. As an expediency, which may or may not be conservative, it was assumed that the cladding has the same fracture-toughness properties as the base material (Eqs. 5.1, 5.2, 5.5, and 5.6). In the OCA-P analysis, assumptions regarding the fracture behavior of the cladding influence only the initiation of very shallow flaws that initially terminate in the cladding. Under some circumstances, including the above assumption regarding the fracture toughness of the cladding, these shallow flaws will initiate and result in vessel failure. Therefore it was necessary to include the fracture properties of the cladding.

The deterministic fracture-mechanics model described above is used in OCA-P to predict the behavior of a flaw during a specified OCA at a specified time in the life of the vessel, and the calculated behavior can be illustrated with a set of critical-crack-depth curves similar to those shown in Figure 5.4. The figure consists of a plot of crack depths corresponding to various events and conditions as a function of the time in the transient at which the events or conditions take place or exist. Figure 5.4 includes (for 2-D flaws only) the locus of points for $K_I = K_{Ic}$ (crack-initiation curve), $K_I = K_{Ia}$ (crack-arrest curve), $K_I = (K_I)_{max}$ (warm prestress curve with $\dot{K}_I = 0$), and K_I = constant (iso K_I curves). For times less than those indicated by the WPS curve, crack initiation will take place, but for greater times initiation will not take place unless perhaps there is a perturbation in K_I that negates the requisite conditions for WPS.

The dashed lines in Figure 5.4 indicate the behavior of two initially shallow flaws, ignoring the effects of WPS. The deeper flaw would initiate at a time of 42 min into the transient and would extend through the wall without arresting. The other flaw would initiate at an earlier time, would arrest at a point 36% of the way through the wall, and then would reinitiate at a time of $\sim$88 min and penetrate the wall. Earlier in the life of the vessel the tendency for complete penetration of the wall is less.

5.3.2 Stress Analysis Model

Using the superposition technique in combination with influence coefficients, the stresses required for the calculation of K_I are those at the crack plane calculated in the absence of the crack and with no variation in the direction of the length of the crack. For the Oconee-1 analysis, it was assumed that there was no azimuthal variation as well, and thus the one-dimensional stress analysis model incorporated in OCA-P was adequate.

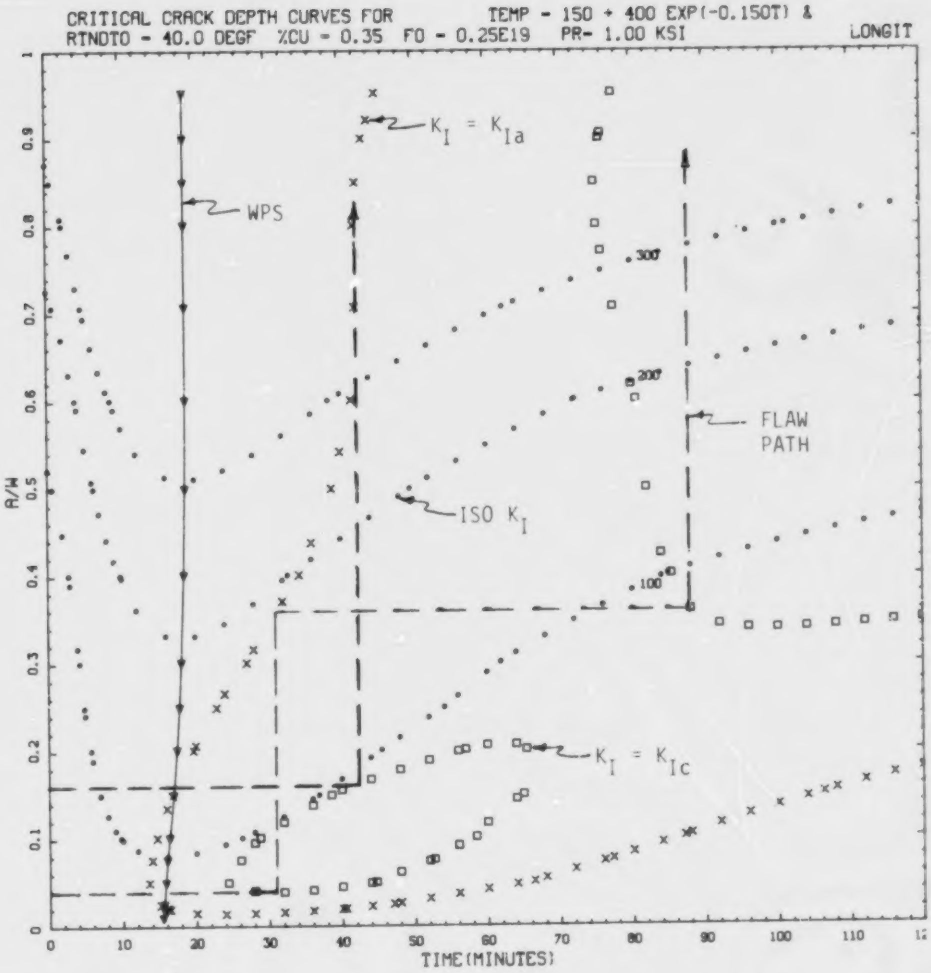

Figure 5.4. Critical-crack-depth curves for a typical postulated overcooling accident.

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Material properties required for the stress analysis included the coefficient of thermal expansion (α), Young's modulus (E), and Poisson's ratio (ν). Although these properties have some temperature dependence, it was determined[8] that the use of appropriate average values results in an error in the calculated value of K_I of less than 10%. Thus, average values were used based on the data in Reference 9. The values used for the Oconee-1 analysis are as follows:

Property	Base Material	Cladding
$\bar{\alpha}$, °C^{-1}	1.45×10^{-5}	1.79×10^{-5}
$\bar{E}$, MPa	1.93×10^{5}	1.86×10^{5}
$\bar{\nu}$	0.30	0.30

5.3.3 Thermal Analysis Model

Temperatures in the wall of the vessel are required for two purposes: to calculate the thermal stresses and to calculate the fracture toughness. The temperatures required for determining the fracture toughness are those in the plane of the flaw, while those used in the one-dimensional analysis of the thermal stresses must represent some type of average distribution through the wall. The thermal stresses in the vicinity of the crack plane are more sensitive to the radial temperature distribution at the crack plane than elsewhere. Since these temperatures are the same as those needed for the fracture-toughness determinations, and since only one set of temperatures was to be used for both the stress and toughness calculations, the local temperatures would be the choice. These particular temperatures were not available, but fortunately the results of the thermal-hydraulic analysis indicated that for the transients of interest there was not much azimuthal variation in the downcomer coolant temperature. Thus, the time-dependent temperature distributions in the wall of the vessel were calculated with the one-dimensional thermal-analysis model in OCA-P using average downcomer coolant temperatures and heat transfer coefficients.

Material properties required for the thermal analysis include the thermal conductivity (k), specific heat (c_p), and density (ρ) of the vessel material. The values used are as follows:

Property	Base Material	Cladding
k, W/m·°C	41.5	17.3
c_p, J/kg·°C	502	502
ρ, Kg/m^3	7830	7830

5.3.4 Probabilistic Analysis Model

The OCA-P probabilistic model, which is similar to that developed by Gamble and Strosnider,[10] is based on Monte Carlo techniques; that is, a large number of vessels is generated, and each vessel is then subjected to a fracture-mechanics analysis to determine whether the vessel will fail. Each vessel is defined by randomly selected values of several parameters that are judged to have significant uncertainties associated with them. The calculated probability of vessel failure is simply the number of vessels that fail divided by the total number of vessels generated. It constitutes a conditional probability of failure, $P(F|E)$, because the assumption is made that the OCA (event) takes place. A logic diagram summarizing the various steps in the OCA-P probabilistic analysis is shown in Figure 5.5.

The parameters simulated for the Oconee-1 analysis are crack depth (a), F_o, RTNDT, Cu, Ni, K_{Ic}, and K_{Ia}. Normal distributions were assumed for all of these parameters except the crack depth; the standard deviations and truncation values used in the analysis are included in Table 5.1.

The probability of having a flaw in a specific weld with a depth in a specific range of crack depths Δa_i is given by

$$P(\Delta a_i) = NV \int_{\Delta a_i} f(a)B(a)da \tag{5.8}$$

where

N = flaws of all depths per unit volume of the specific weld

V = volume of the specific weld

f(a) = flaw-depth density function

B(a) = probability of nondetection

The parameters N and f(a) pertain to vessel conditions prior to preservice inspection and repair, and B(a) is derived on the basis of repairing or otherwise disposing of all detected flaws.

The value of N and the functions f(a) and B(a) are not well known because most of the available inspection data do not pertain to surface flaws that extend into and through the cladding of a PWR pressure vessel. For the Oconee-1 analysis, the functions f(a) and B(a) were those suggested in the Marshall Report[11] and are as follows:

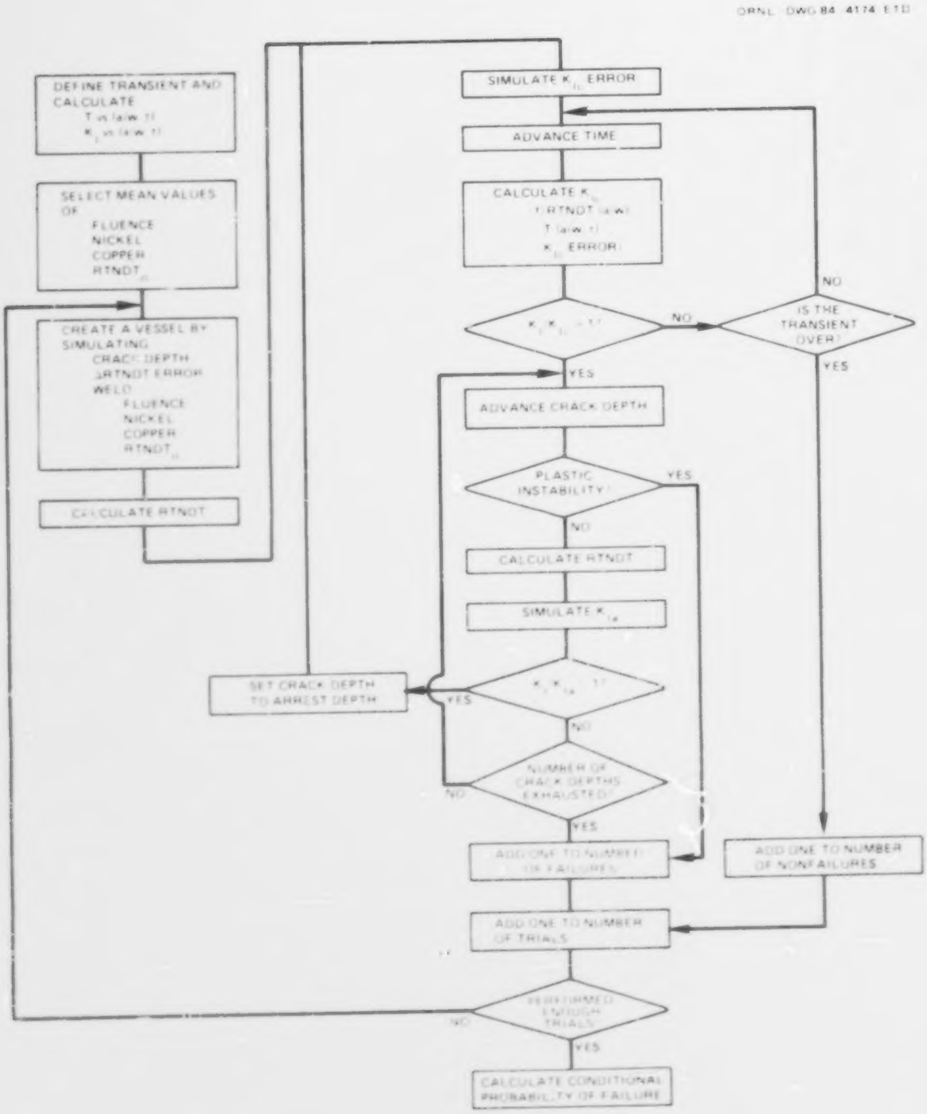
ORNL DWG 84 4174 ETD
DEFINE TRANSIENT AND CALCULATE T vs (a/w, t) K_I vs (a/w, t)
SELECT MEAN VALUES OF FLUENCE NICKEL COPPER RTNDT_0
CREATE A VESSEL BY SIMULATING CRACK DEPTH ΔRTNDT ERROR WELD FLUENCE NICKEL COPPER RTNDT_0
CALCULATE RTNDT
SIMULATE K_Ic ERROR
ADVANCE TIME
CALCULATE K_Ic f(RTNDT (a/w), T (a/w, t), K_Ic ERROR)
K_I/K_Ic ≥ 1?
NO
IS THE TRANSIENT OVER?
NO
YES
YES
ADVANCE CRACK DEPTH
PLASTIC INSTABILITY?
YES
NO
CALCULATE RTNDT
SIMULATE K_Ia
SET CRACK DEPTH TO ARREST DEPTH
K_I/K_Ia < 1?
YES
NO
NUMBER OF CRACK DEPTHS EXHAUSTED?
NO
YES
ADD ONE TO NUMBER OF FAILURES
ADD ONE TO NUMBER OF NONFAILURES
ADD ONE TO NUMBER OF TRIALS
PERFORMED ENOUGH TRIALS?
NO
YES
CALCULATE CONDITIONAL PROBABILITY OF FAILURE

Figure 5.5. OCA-P program logic.

Table 5.1. Parameters simulated in OCA-P

Parameter	Standard* Deviation (σ)	Truncation
Fluence (F)	0.3 $\mu(F)$	F = 0
Copper	0.025%	--
Nickel	0.0	--
$RTNDT_o$	9°C**	**
$\Delta RTNDT$	13°C**	**
K_{Ic}	0.15 $\mu(K_{Ic})$	$\pm 3\sigma$
K_{Ia}	0.10 $\mu(K_{Ia})$	$\pm 3\sigma$

*Normal distribution used for each parameter.

** $\sigma_{(RTNDT)} = \left[\sigma^2_{(RTNDT_o)} + \sigma^2_{(\Delta RTNDT)}\right]^{1/2}$, truncated at $\pm 3\sigma$.

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$$f(a) = 0.16\ e^{-0.16\ a} \tag{5.9}$$

$$B(a) = 0.005 + 0.995\ e^{-0.113\ a} \tag{5.10}$$

where

a = crack depth, mm

$\int_0^\infty f(a)da = 1$

For the Oconee-1 vessel the probability of nondetection, B(a), should probably be set equal to unity, independent of a, because it is not likely that a reliable inspection was made for flaws in and extending a short distance beyond the cladding. Furthermore, it is not likely that any detected flaws of this type were repaired. Even so, Eq. 5.1 was used in the Oconee-1 analysis. If B(a) = 1 were used instead, $P(F|E)$ would be about twice as much. Thus the results of this study can be interpreted accordingly.

After Oconee-1 had been in operation for ~10 years, an in-service inspection of the reactor pressure vessel was performed. No flaws were found; however, the significance of this inspection in terms of differences between the assumed and actual flaw-depth density function and flaw density was not apparent at the time of this writing.

The value of N used in the Oconee-1 analysis was 1 flaw/m^3 of weld material, and it was assumed that all flaws were inner-surface flaws normal to the surface and oriented in the length-direction of the weld.

This value of the flaw density agrees with that suggested in the Marshall Report, but the uncertainty is considered to be very large ($\sigma \simeq 10^2$).

The volume (V) of a weld used for calculating the number of surface flaws in a weld was the total volume of that portion of the weld that was nearly within the axial confines corresponding to the active length of the core.

As mentioned above, the calculated probability of vessel failure for this study is the number of simulated vessels calculated to fail divided by the total number of vessels simulated or otherwise accounted for. Thus,

$$P(F|E) = \sum_j \frac{N'_{fj}}{N'_{vj}} V_j N \int_0^w f(a)B(a)da \qquad (5.11)$$

where

N'_{fj} = number of vessels with a flaw in the jth weld that fail

N'_{vj} = number of vessels simulated with a flaw in the jth weld

V_j = volume of jth weld

The integral in Eq. 5.11 accounts for the vessels that have no flaws whatsoever, and each term in Eq. 5.11 represents the contribution to P(F|E) of each weld.

For very small values of P(F|E), the value of N'_{vj} required to achieve reasonable accuracy becomes quite large. Under some circumstances the value of N'_{vj} can be reduced by using stratified sampling of one or more of the parameters simulated. This was done for the flaw depth, assuming a uniform distribution of depths. This procedure allows a more frequent sampling of the less probable deep flaws, which, for low-probability transients that are characterized by high pressure and a mild thermal shock, are responsible for most of the initiation events that lead to failure. The results are then weighted by the actual flaw-depth density to obtain

$$P(F|E) = \sum_j \sum_j \frac{N'_{fij}}{N'_{vij}} \left[\frac{\int_{\Delta a_i} f(a)B(a)da}{\int_0^W f(a)B(a)da} \right] NV_j \int_0^W f(a)B(a)da \tag{5.12}$$

where

N'_{fij} = number of vessels that fail with a flaw in the jth weld with depth in Δa_i

N'_{vij} = number of vessels simulated with a flaw in the jth weld with depth in Δa_i

A deterministic analysis is made for each of the simulated vessels to determine if failure will occur during a particular transient at a specified time in the life of the plant. The criterion by which failure is judged is as follows: if, following an initiation event, K_I remains greater than K_{Ia} up to or beyond the point at which plastic instability occurs in the remaining ligament, failure is assumed. The onset of plastic instability is evaluated on the basis of achieving an average pressure stress in the remaining ligament equal to the flow stress. The flow stress is assumed to be independent of temperature and fluence and is specified as 550 MPa.

The number of vessels that must be simulated depends upon the accuracy required for the calculated value of $P(F|E)$, and as small a number as practical is used to minimize computer costs. The minimum number of simulated vessels required to satisfy a specified accuracy is estimated by applying the central limit theorem.[12] Using this approach and specifying a 95% confidence level yields

$$P(F|E)_j = \hat{P}_j \, NV_j \int_0^W f(a)B(a)da \pm 1.96 \, \sigma_j \tag{5.13}$$

where

$P(F|E)_j$ = true value of the conditional probability of vessel failure for those vessels having flaws in the jth weld only

σ_j = one standard deviation

$$\hat{P}_j = \frac{N'_{fj}}{N'_{vj}}$$

For the direct approach (not using stratified sampling),

$$\sigma_j = \left[\frac{\hat{P}_j(1 - \hat{P}_j)}{N'_{vj}}\right]^{1/2} NV_j \int_0^W f(a)B(a)da \tag{5.14}$$

When stratified sampling is used,

$$\sigma_j = \left\{ \sum_i \left[\frac{\int_{\Delta a_i} f(a)B(a)da}{\int_0^W f(a)B(a)da}\right]^2 \left[\frac{\hat{P}_{ij}(1 - \hat{P}_{ij})}{N'_{vij}}\right] \right\}^{1/2} NV_j \int_0^W f(a)B(a)da \tag{5.15}$$

where

$$\hat{P}_{ij} = \frac{N'_{fij}}{N'_{vij}}$$

The value of σ corresponding to all of the vessels simulated is

$$\sigma_{P(F|E)} = \sqrt{\sum_j \sigma_j^2} \tag{5.16}$$

and the error, ε_j, associated with the jth weld is

$$\varepsilon_j = \frac{1.96\ \sigma_j}{\hat{P}_j\ NV_j \int_0^W f(a)B(a)da} \tag{5.17}$$

The total error, ε, considering all welds of interest is

$$\varepsilon = \frac{1.96\ \sigma_{P(F|E)}}{\sum_j \hat{P}_j\ NV_j \int_0^W f(a)B(a)da} \tag{5.18}$$

Three specific criteria were used in selecting the number of vessels to be simulated:

(1) $(N'_{vj})_{max} = 500{,}000$

(2) $(N'_{vj})_{min} = 10{,}000$

(3) $\varepsilon_j = 10\%$

The application of these criteria in terms of ε_j vs $\hat{P}_j$ is shown in Figure 5.6 for the direct (nonstratified) sampling method.

For Oconee-1, the quantity $NV_j \int_0^W f(a)B(a)da$ varies from 0.6×10^{-2} to 2.3×10^{-2}. Thus, within a factor of about 2, $P(F|E)_j$ is two orders of magnitude less than $\hat{P}_j$, and the information in Figure 5.6 can be interpreted accordingly. For instance, suppose (1) there is a need to calculate values of $P(F|E)$ as low as 3×10^{-7}, (2) there are three welds of concern, and (3) a corresponding value of $P(F|E)_j$ is 1×10^{-7}. Thus, $\hat{P}_j \simeq 1 \times 10^{-5}$, and from the curve in Figure 5.6, $\varepsilon_j \simeq 80\%$. The error in $P(F|E)$ would be approximately $80\%/\sqrt{3} \simeq 50\%$.

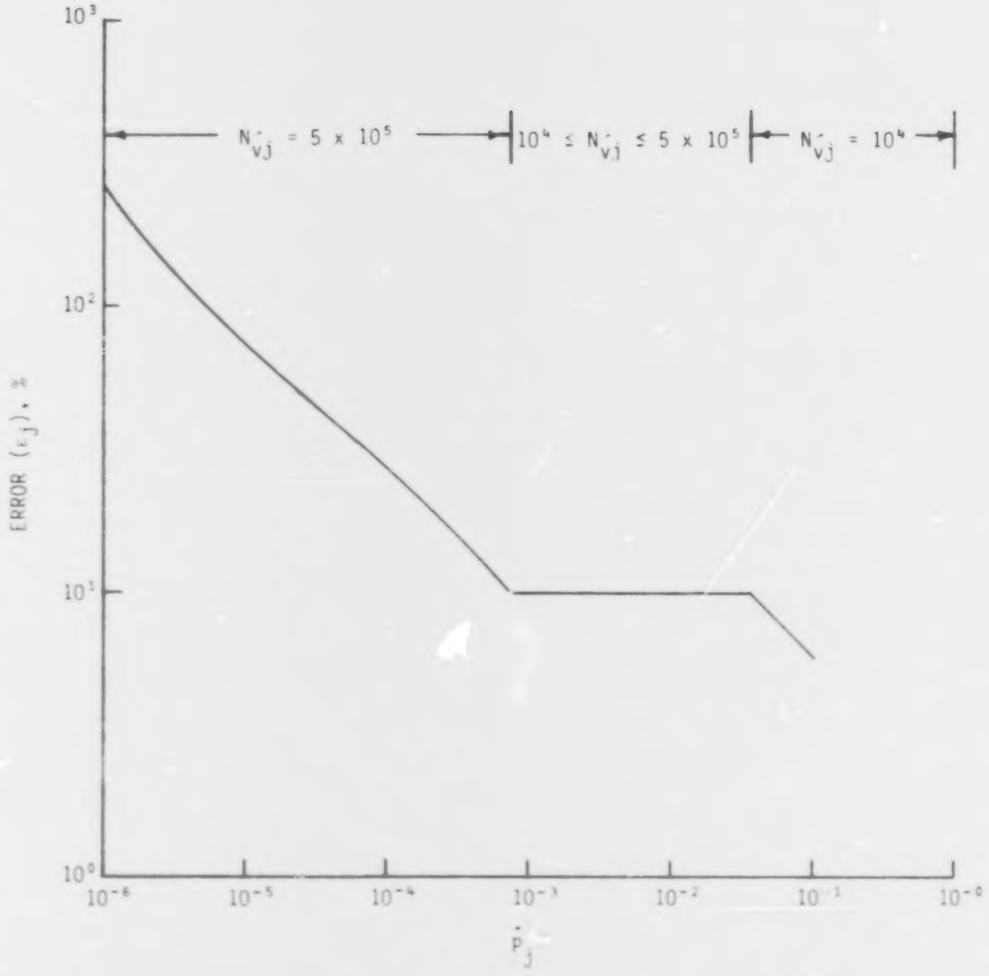

Figure 5.6. Graphic illustration of the error in $\hat{P}_j$, consistent with the criteria used for establishing the number of vessels simulated (N'_{vj}).

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For the purpose of estimating the absolute frequency of vessel failure or identifying dominant transients, the magnitude of the errors indicated in Figure 5.6 is acceptable. However, for the sensitivity studies, larger values of N'_{vj} were used where appropriate to reduce the error.

5.4 Flaw-Related Data for the Oconee-1 Reactor Pressure Vessel

As already mentioned, the areas of the vessel of particular concern with regard to flaw propagation are the ones that are most likely to have flaws and relatively high values of F_o, $RTNDT_o$, Cu, and Ni. The region directly opposite the active portion of the core is exposed to the highest neutron fluxes, and the attenuation beyond the active length of the core is very steep. Thus, only this beltline region of the vessel was considered.

Within the beltline region the concentration of copper is significantly less in the base material than in the welds, as indicated by the data in Table 5.2 and Figures 5.2, 5.7, and 5.8. Preliminary OCA-P calculations (see Appendix D) indicated that because of its lower copper concentration the base material contributed relatively little to vessel failure, assuming the same flaw density in both the base material and welds. These preliminary calculations also indicated that welds SA1073, 1430, and 1493, each of which is oriented in an axial direction, contributed far more than all the other welds. Furthermore, the staggered arrangement of these six welds (two each) and the relatively low concentration of copper in the base material tend to prevent surface extension of a

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Table 5.2. Material properties used in the LEFM analysis of the Oconee-1 reactor vessel

Material Identification*		Chemistry** (wt %)		Neutron Fluence, Inside Surface (n/cm^2)		Initial*** RTNDT (°C)	Volume (m^3)
Weld or Heat Number	Type	Cu	Ni	5 EFPY	32 EFPY		
AHR-54	SA508, CL2	0.16	NA		1.97E18	(+16)	
SA1135	Circumferential weld	0.25	0.54		1.97E18	(-7)	
C2197	SA302B	0.15	NA		9.35E18	(+4)	
SA1073	Longitudinal weld	0.31	0.64	1.43E18	7.38E18	(-7)	0.01
SA1229	Circumferential weld	0.26	0.61		9.35E18	(-7)	
SA1493	Longitudinal weld	0.29	0.55	1.74E18	8.98E18	(-7)	0.04
C3278-1	SA302B	0.12	NA		1.23E19	(+4)	
SA1585	Circumferential weld	0.21	0.59		1.23E19	(-7)	
C2800-1	SA302B	0.11	NA		1.23E19	(+4)	
SA1430	Longitudinal weld	0.29	0.55	2.11E18	1.09E19	(-7)	0.04

*Source: BAW-1436, September 1977.

**Chemistry source: BAW-1511P, October 1980.

***Estimated RTNDT values source: BAW-10046A, Rev. 1, March 1976.

NA -- Not available.

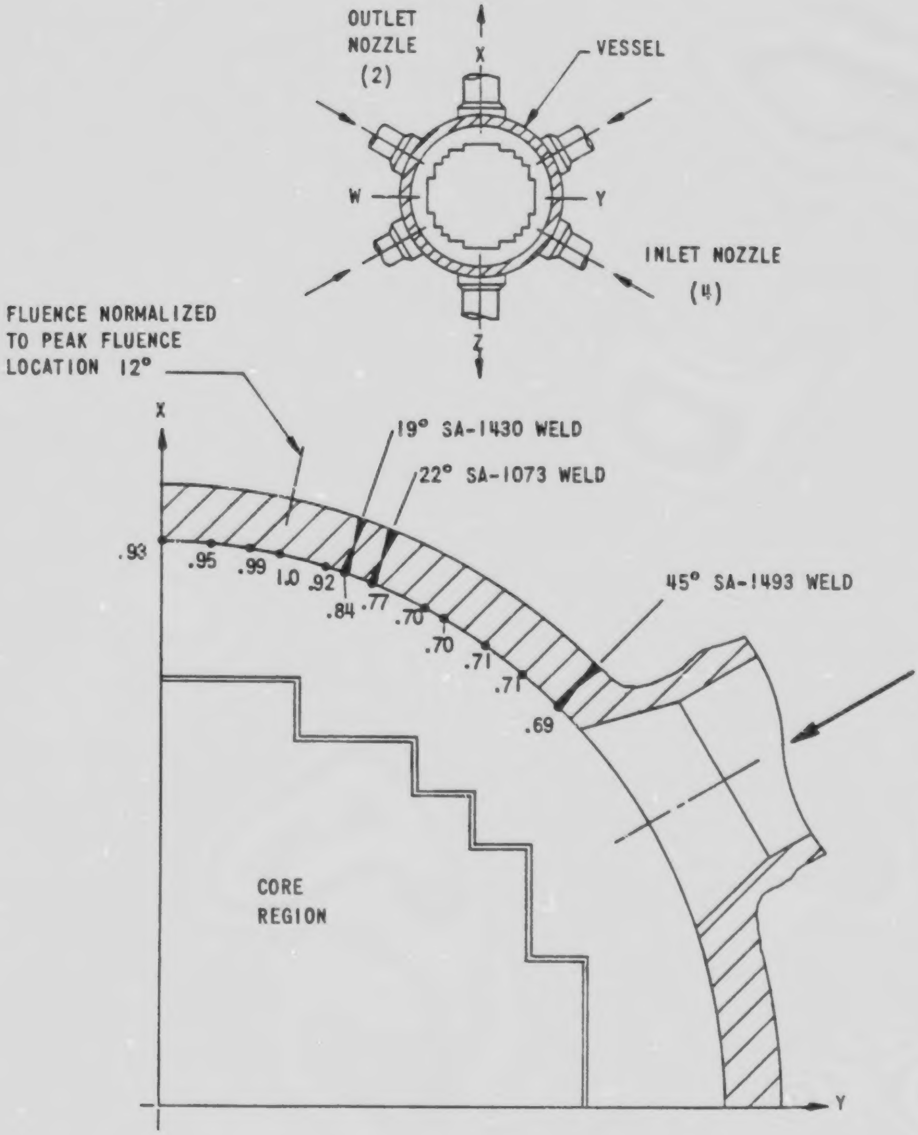
OUTLET NOZZLE (2)
X
VESSEL
W
Y
INLET NOZZLE (4)
Z
FLUENCE NORMALIZED TO PEAK FLUENCE LOCATION 12°
X
19° SA-1430 WELD
22° SA-1073 WELD
.93
.95
.99
1.0
.92
.84
.77
.70
.70
.71
.71
.69
45° SA-1493 WELD
CORE REGION
Y

Figure 5.7. Longitudinal weld locations relative to azimuthal fluence profile.[13]

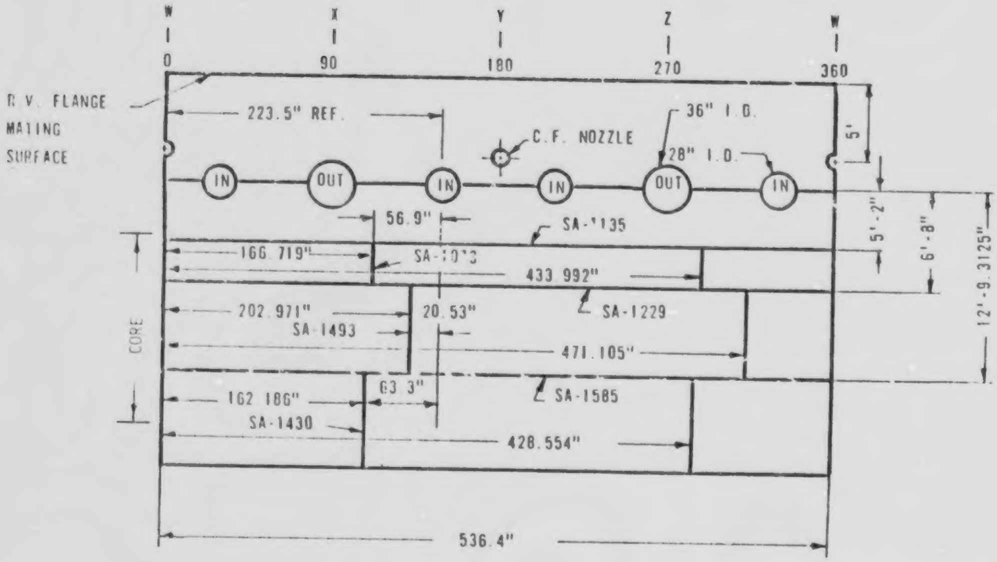
W
X
Y
Z
W
0
90
180
270
360
R.V. FLANGE MATING SURFACE
223.5" REF.
C.F. NOZZLE
36" I.D.
28" I.D.
IN
OUT
IN
IN
OUT
IN
5'
56.9"
SA-1135
5'-2"
6'-8"
12'-9.3125"
166.719"
SA-1073
433.992"
202.971"
20.53"
SA-1229
SA-1493
471.105"
CORE
162.186"
63.3"
SA-1585
SA-1430
428.554"
536.4"

Figure 5.8 Developed view of inner surface of Oconee-1 reactor vessel showing weld locations.[13]

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flaw beyond the ends of the welds. Thus, the only regions of the vessel considered in the final analysis were these six welds.

Values of Cu, Ni, F_o, $RTNDT_o$, and V for the six welds of concern are included in Table 5.2. The fluence rate, F/EFPY, was assumed to be constant before and after 5 EFPY, at which time the fluence rate at Oconee was reduced. The weld volumes are based on the total lengths of the welds because, as shown in Figure 5.2, the uppermost weld (SA1073) is fully within the active length of the core and the lower weld (SA1430) extends only a short distance beyond the active length.

5.5 Results of Analysis

5.5.1 Types of Analyses Conducted

Probabilistic fracture-mechanics calculations were performed to determine (1) the conditional probability of vessel failure [$P(F|E)$] for a number of postulated Oconee-1 transients, (2) the sensitivity of $P(F|E)$ to small changes in the mean values of certain parameters, (3) the effect of including WPS, and (4) the effect on $P(F|E)$ of certain proposed remedial measures. The results of these efforts are presented below.

5.5.2 Conditional Probability of Vessel Failure

The specific transients considered for a detailed OCA-P analysis are described in Chapter 3, and the ones actually calculated are indicated in Table 5.3. Those transients not calculated were judged to have values of P(F|E) at 32 EFPY less than 10^{-7}.

For those transients calculated, values of P(F|E) were determined for several times (EFPYs) in the expected life of the vessel. The results are summarized in Table 5.3 and are presented graphically in Figure 5.9 as P(F|E) vs EFPY. Also included on the abscissa of Figure 5.9 are the mean values of RTNDT and F at the inner surface of the vessel for weld SA1430.

A summary of more detailed results for each transient calculated is presented in digital form in Appendix E. An example is shown in Table 5.4 for transient No. 44. The summary sheet includes the data for a variety of histograms; a set of four histograms for transient No. 44 is shown in Figures 5.10--5.13.

Appendix E also includes for each transient calculated a definition of the transient input to OCA-P (downcomer coolant temperature vs time, primary-system pressure vs time, and fluid-film heat transfer coefficient at the vessel inner surface vs time), temperature distributions in the wall, and a set of critical-crack-depth curves for weld SA1430 based on mean values of all parameters except K_{Ic} and K_{Ia}, which are -2σ values. Examples of these graphical outputs are shown in Figures 5.14--5.17 for transient No. 44.

Table 5.3. Summary of calculated values of $P(F \mid E)$ for the Oconee-1 postulated transients

Transient*	Conditional Probability of Failure, $P(F \mid E)$					
	EFPY					
	3.9	6.9	15.3	23.6	32.0	42.0
	F_o,** 10^{19} neutrons/cm^2					
	0.164	0.273	0.545	0.818	1.090	1.417
	RTNDT,** °C					
	55	65	79	89	97	104
1				2×10^{-6}	3×10^{-7}	1.7×10^{-6}
1A, 5, 7				1.1×10^{-6}	5.2×10^{-6}	1.6×10^{-5}
4, 9	1.2×10^{-5}	6.0×10^{-5}	4.0×10^{-4}	1.1×10^{-3}	2.0×10^{-3}	3.4×10^{-3}
6				3×10^{-7}	1.7×10^{-6}	4.9×10^{-6}
10				3.1×10^{-6}	9.8×10^{-6}	3.1×10^{-5}
14	1.2×10^{-5}	4.8×10^{-5}	2.8×10^{-4}	7.0×10^{-4}	1.3×10^{-3}	2.0×10^{-3}
21				1×10^{-7}	9×10^{-7}	5.2×10^{-6}
24		4×10^{-7}	8.8×10^{-6}	3.8×10^{-5}	1.0×10^{-4}	2.2×10^{-4}
25		8.4×10^{-6}	7.8×10^{-5}	2.4×10^{-4}	5.2×10^{-4}	9.8×10^{-4}
26, 27, 32		2.2×10^{-6}	5.0×10^{-5}	2.4×10^{-4}	6.2×10^{-4}	1.3×10^{-3}
28, 30		2.4×10^{-6}	4.2×10^{-5}	1.6×10^{-4}	4.0×10^{-4}	8.6×10^{-4}
29	4.2×10^{-6}	3.8×10^{-5}	4.2×10^{-4}	1.4×10^{-3}	3.0×10^{-3}	5.4×10^{-3}
33		1.2×10^{-5}	1.0×10^{-4}	3.2×10^{-4}	6.5×10^{-4}	1.2×10^{-3}
34		8.3×10^{-5}	4.4×10^{-4}	1.1×10^{-3}	1.8×10^{-3}	2.9×10^{-3}
41	7.4×10^{-6}	4.2×10^{-5}	3.2×10^{-4}	9.0×10^{-4}	1.8×10^{-3}	3.0×10^{-3}
44	1.3×10^{-4}	4.0×10^{-4}	1.7×10^{-3}	3.6×10^{-3}	5.4×10^{-3}	8.0×10^{-3}
46				1×10^{-7}	1.0×10^{-6}	4.4×10^{-6}
57			6.1×10^{-6}	4.5×10^{-5}	1.5×10^{-4}	4.1×10^{-4}

*Refer to Table 6.1 for correlation of transient number with definition of transient.

**Mean values for weld SA1430.

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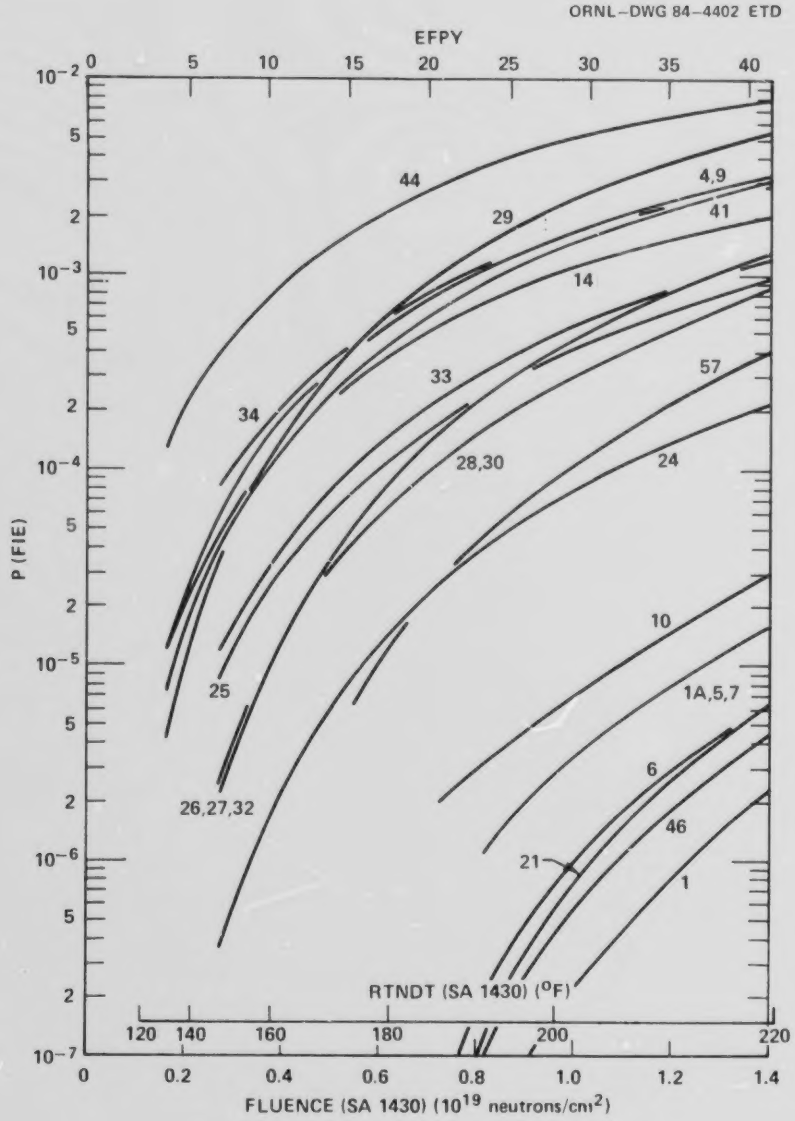

Figure 5.9. P(F|E) vs EFPY for Oconee-1 postulated OCAs.

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Table 5.4. Summary sheet of OCA-P output for Oconee-1 postulated transient No. 44

```
IPTS OCONEE CLAD 6A  10/20/83                               1. FLAWS/M**3      FO =  1.090D+19

       -------------UNADJUSTED--------------               ---ADJUSTED----
WELD    P(F/E)       95%CI     %ERR   P(INITIA)    N*V      P(F/E)     %ERR     NTRIALS


 1      6.74D-02   3.67D-03    5.45   6.75D-02   0.040    2.69D-03                10000
 2      5.09D-02   3.24D-03    6.37   5.11D-02   0.040    2.03D-03                10000
 3      7.56D-02   3.86D-03    5.10   7.61D-02   0.010    7.56D-04                10000

                                                 VESSEL    5.49D-03   3.64



 DEPTHS FOR INITIAL INITIATION (MM)
                 3.2    6.3   12.7   19.0   25.4   38.1   50.8
   NUMBER       1033   1614    501    137     26      3      0
   PERCENT      31.2   48.7   15.1    4.1    0.8    0.1    0.0



TIMES OF FAILURE(MINUTES)
                   0.0   10.0   20.0   30.0   40.0   50.0   60.0   70.0   80.0   90.0 100.0 110.0 120.0
   NUMBER                 0      0      0      0      2     17     49    114    282    535  1006  1296
   PERCENT              0.0    0.0    0.0    0.0    0.1    0.5    1.5    3.5    8.5   16.2  30.5  39.3



 INITIATION T-RTNDT(DEG.C)
                 -55.6 -41.7 -27.8 -13.9    0.0   13.9   27.8   41.7   55.6   69.4   83.3   97.2 111.1
   NUMBER                74    284    969   1252    649     83      3      0      0      0      0      0
   PERCENT              2.2    8.6   29.2   37.8   19.6    2.5    0.1    0.0    0.0    0.0    0.0    0.0



 ARREST T-RTNDT(DEG.C)
                 -27.8 -13.9    0.0   13.9   27.8   41.7   55.6   69.4   83.3   97.2 111.1 125.0 138.9
   NUMBER                 0      0      6      7      0      0      0      0      0      0      0      0
   PERCENT              0.0    0.0   46.2   53.8    0.0    0.0    0.0    0.0    0.0    0.0    0.0    0.0
```

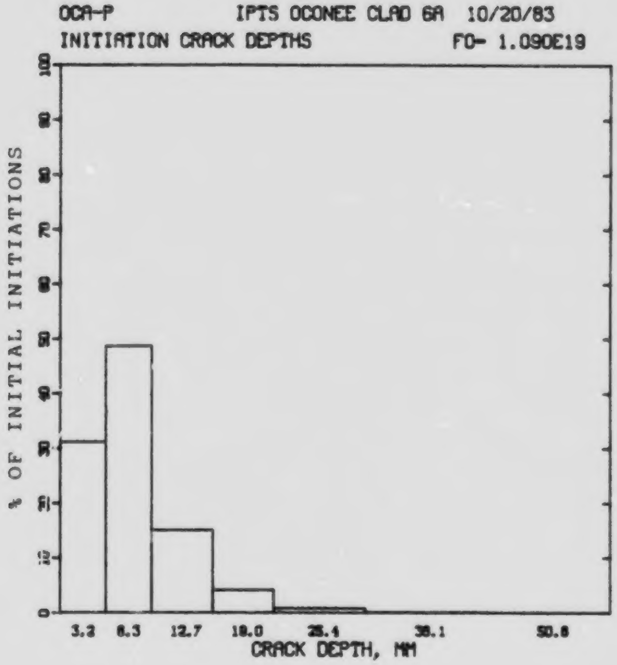

Figure 5.10 Histogram of percent initiations vs crack depth for first initiation event (Oconee-1 postulated transient No. 44; at 32 EFPY).

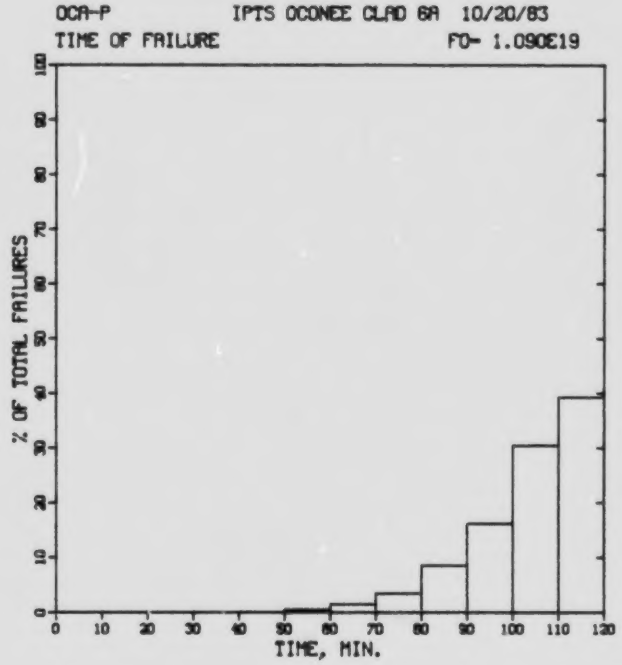

Figure 5.11. Histogram of percent failures vs time of failure (Oconee-1 postulated transient No. 44; at 32 EFPY).

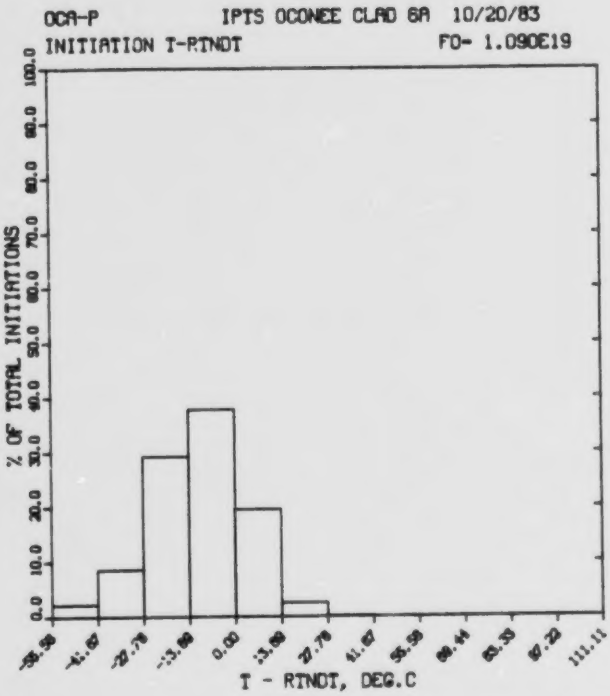

Figure 5.12 Histogram of percent initiations vs T - RTNDT (weld SA1430) at tip of flaw for all initiation events (Oconee-1 postulated transient No. 44; 32 EFPY).

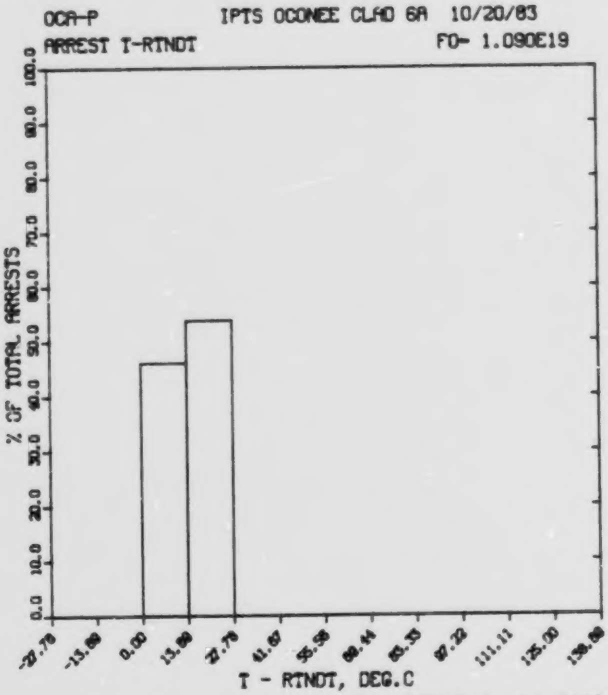

Figure 5.13. Histogram of percent arrests vs T - RTNDT (weld SA1430) at tip of flaw for all arrest events (Oconee-1 postulated transient No. 44; 32 EFPY).

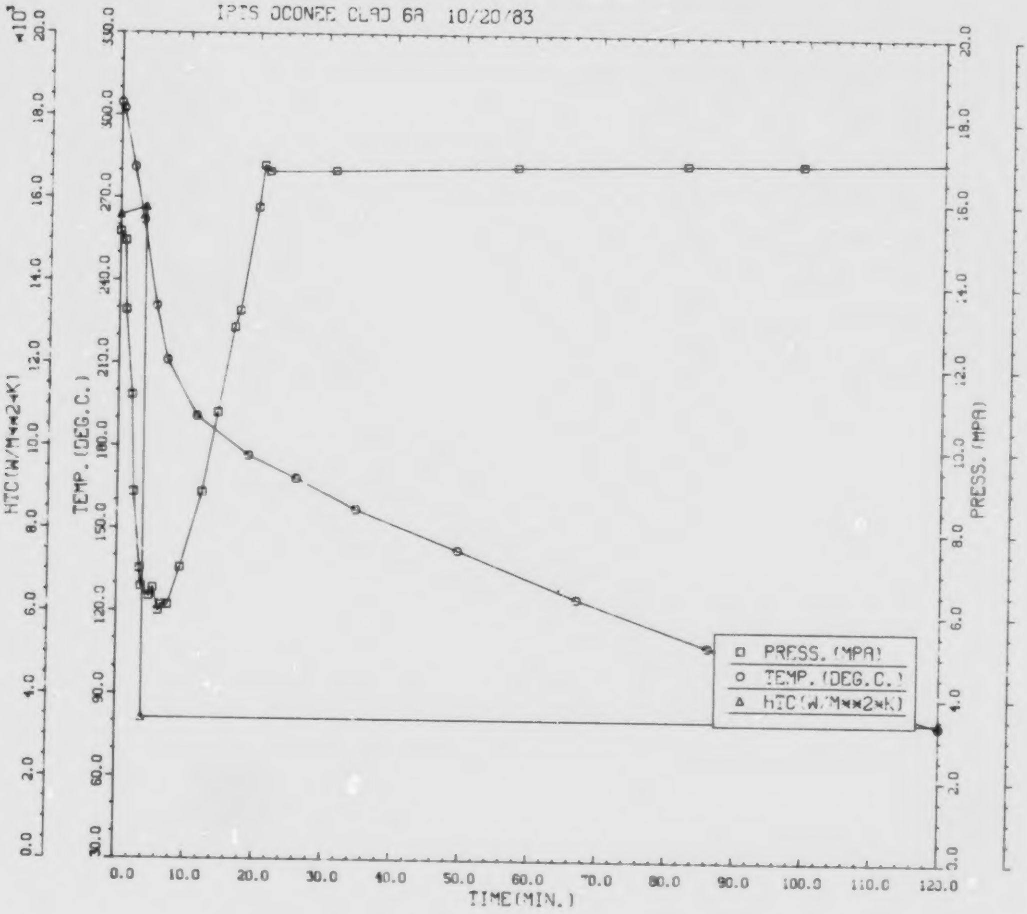

Figure 5.14. P, T, and h vs t for Oconee-1 postulated transient No. 44.

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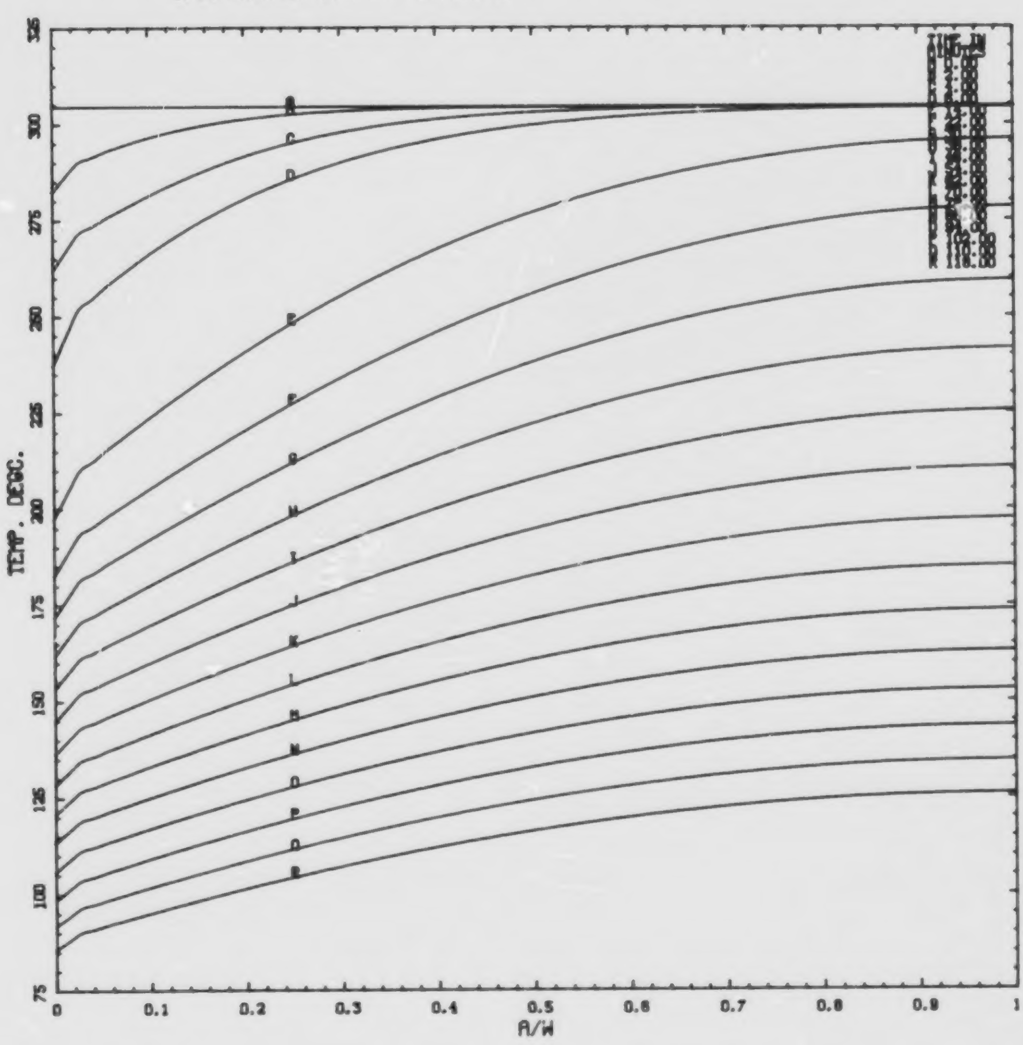

Figure 5.15. Wall temperature vs a/w, t for Oconee-1 postulated transient No. 44.

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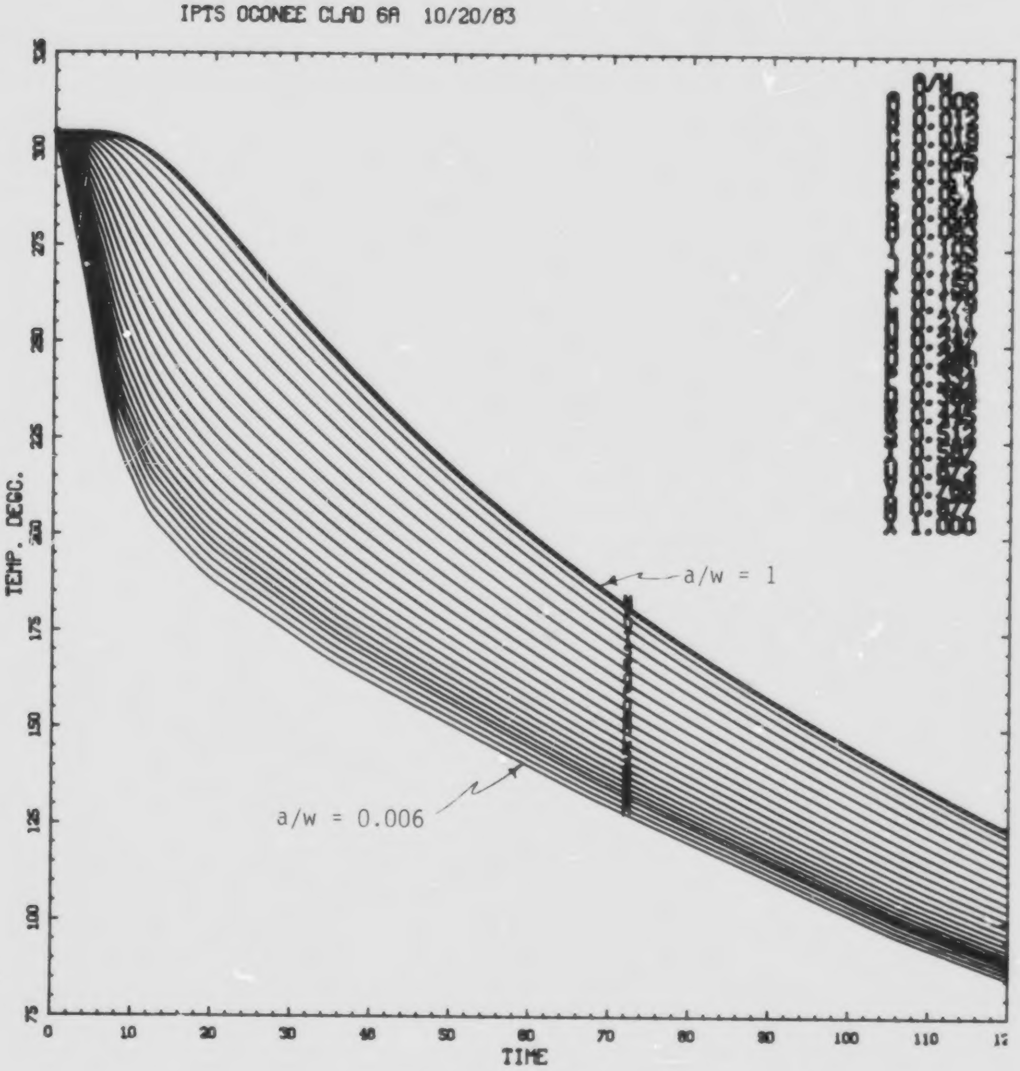

Figure 5.16. Wall temperature vs t, a/w for Oconee-1 postulated transient No. 44.

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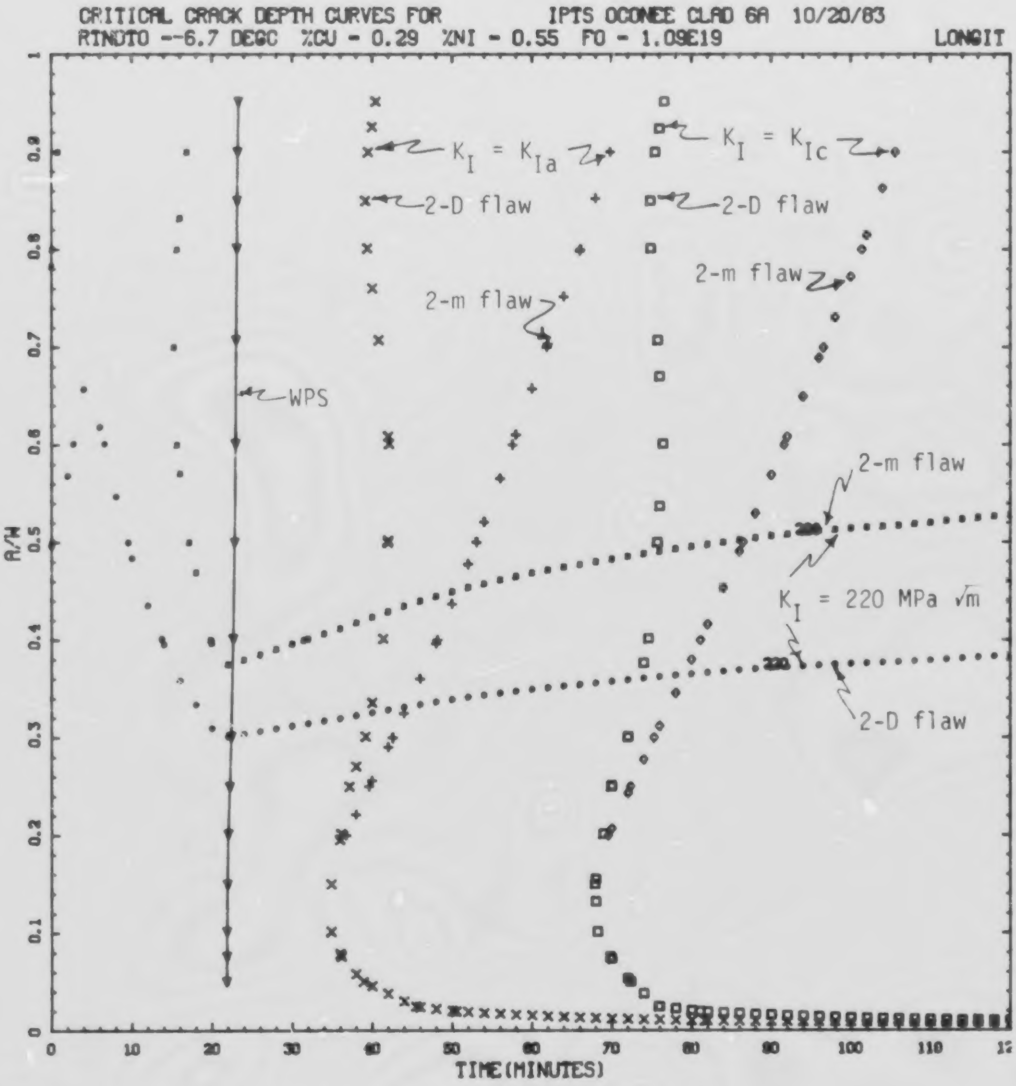

Figure 5.17. Critical-crack-depth curves for Oconee-1 postulated transient No. 44, 32 EFPY, weld SA1430.

5.5.3 Sensitivity Analysis

The sensitivity analysis was conducted by determining the change in $P(F|E)$ corresponding to a change in the mean value of each of several parameters. The mean value of only one parameter was changed at a time while all other parameters retained their original mean values. The parameters changed were K_{Ic}, K_{Ia}, RTNDT, Cu, F, fluid-film heat transfer coefficient, downcomer coolant temperature, primary-system pressure, and flaw density. The amount of the change for each parameter was one standard deviation, and for some parameters both plus and minus changes were made because of a suspected strong nonlinearity.

The values of σ used in the sensitivity analysis for K_{Ic}, K_{Ia}, RTNDT, Cu, and F are listed in Table 5.1, and the values of the flaw density, N, corresponding to the application of $\pm 1\sigma$ were 10^2 and 10^{-2} times the original mean value. The 1σ change in the downcomer coolant temperature constituted a linear change in temperature from zero at time zero to 28°C at a time corresponding to the minimum point in the temperature vs time curve. From then on, the change in temperature was a constant value of 28°C. The 1σ change in the heat transfer coefficient, h, was 0.25 h and for the pressure, 0.34 MPa.

The results of the sensitivity study are presented in Table 5.5 for 32 EFPY. Table 5.5 includes the values of $P(F|E)$ corresponding to the original mean values of the parameters and the changes in $P(F|E)$ corresponding to each of the 1σ changes.

It is of interest to note (bottom of Table 5.5) that aside from the sensitivity to N, $P(F|E)$ is most sensitive by far to the reduction in downcomer coolant temperature and is about a factor of 6 less sensitive to a reduction in K_{Ic} and an increase in RTNDT. The probability of failure is least sensitive to variations in the heat transfer coefficient and the primary system pressure.

5.5.4 Effect of Including WPS

During many of the postulated OCAs, the stress intensity factor, K_I, for all crack depths first increases with time, reaches a maximum, and then decreases. For the shallow flaws that are generally responsible for the initial crack initiation event, once K_I begins to decrease it does so throughout the remainder of the transient. This time-dependent behavior of K_I may prevent failure of a vessel because a flaw cannot initiate while K_I is decreasing, even though $K_I/K_{Ic} \geq 1$. This phenomenon, which, as noted earlier, is referred to as warm prestressing (WPS), and the time of incipient WPS is the time at which $\dot{K}_I$ becomes equal to zero.

For most of the Oconee-1 postulated transients, WPS could be a factor because the calculations indicate that K_I does not become equal to K_{Ic} until after the time of WPS. A typical case is illustrated in Figures 5.4 and 5.11. The reason for not including WPS in most of the calculations is that the K_I vs t curves for the shallow flaws are very flat, making it difficult to determine where the maximum is. Furthermore, unforeseen perturbations in pressure and coolant temperature might exist and defeat WPS. Even so, it is of interest to see what the effect is

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Table 5.5. Sensitivity of P(F|E) to 1σ changes in the mean values of several of the simulated parameters at 32 EFPY

Transient	P(F|E) original mean values	ΔP(F|E)											
		Simulated Parameter											
		K_{Ic}		K_{Ia}	RTNDT		F_0	Cu		h	T_c		ρ
		+σ	−σ	+σ	+σ	−σ	+σ	+σ	−σ	+σ	−σ	+σ	−σ
1	3×10^{-7}												*
1A, 5, 7	5.2×10^{-6}												*
4, 9	2.0×10^{-3}	-1.3×10^{-3}	3.6×10^{-3}	-5.4×10^{-5}	3.4×10^{-3}	-1.5×10^{-3}	1.4×10^{-3}	1.0×10^{-3}		1.6×10^{-4}	9.9×10^{-3}	1.9×10^{-3}	*
6	1.7×10^{-6}												*
10	9.8×10^{-6}												*
14	1.3×10^{-3}	-9.2×10^{-4}	3.1×10^{-3}	-1.4×10^{-4}	2.2×10^{-3}	-9.2×10^{-4}	7.4×10^{-4}	9.7×10^{-4}		5.4×10^{-5}	7.6×10^{-3}	-1.2×10^{-3}	-2×10^{-4}
21	9×10^{-7}												*
24	1.0×10^{-4}												*
25	5.2×10^{-4}	-3.8×10^{-4}	1.4×10^{-3}	-2.8×10^{-5}	1.3×10^{-3}	-4.1×10^{-4}	4.6×10^{-4}	5.4×10^{-4}		2.8×10^{-5}	5.2×10^{-3}	-4.8×10^{-4}	*
26, 27, 32	6.2×10^{-4}	-4.1×10^{-4}	1.1×10^{-3}	-2.2×10^{-5}	2.7×10^{-3}	-5.6×10^{-4}	7.0×10^{-4}	9.0×10^{-4}	-4.1×10^{-4}	9.0×10^{-5}	1.3×10^{-2}	-6.2×10^{-4}	-1.5×10^{-4}
28, 30	4.0×10^{-4}	-2.0×10^{-4}	4.8×10^{-4}	-7.0×10^{-5}	1.7×10^{-3}	-3.5×10^{-4}	4.6×10^{-4}	5.1×10^{-4}		8.8×10^{-5}	1.0×10^{-2}	4.0×10^{-4}	-1.4×10^{-4}
29	3.0×10^{-3}	-1.8×10^{-3}	4.1×10^{-3}	-4.0×10^{-5}	7.0×10^{-3}	-2.4×10^{-3}	2.4×10^{-3}	2.9×10^{-3}		3.6×10^{-4}	2.1×10^{-2}	-2.9×10^{-3}	*
33	6.5×10^{-4}	-4.8×10^{-4}	1.6×10^{-3}	-5.0×10^{-5}	1.3×10^{-3}	-5.1×10^{-4}	5.5×10^{-4}	6.5×10^{-4}		2.0×10^{-5}	5.1×10^{-3}	6.3×10^{-4}	*
34	1.8×10^{-3}	-1.3×10^{-3}	3.8×10^{-3}	-8.2×10^{-5}	2.7×10^{-3}	-1.3×10^{-3}	1.1×10^{-3}	1.3×10^{-3}		5.0×10^{-5}	9.9×10^{-3}	-1.7×10^{-3}	*
41	1.8×10^{-3}	-1.2×10^{-3}	3.2×10^{-3}	-6.2×10^{-5}	3.2×10^{-3}	-1.3×10^{-3}	1.2×10^{-3}	1.5×10^{-3}	-9.5×10^{-4}	1.1×10^{-4}	1.1×10^{-2}	-1.7×10^{-3}	*
44	5.4×10^{-3}	-3.4×10^{-3}	7.6×10^{-3}	-2.0×10^{-5}	5.9×10^{-3}	-3.5×10^{-3}	2.6×10^{-3}	3.1×10^{-3}	-2.2×10^{-3}	3.6×10^{-4}	1.8×10^{-2}	-5.1×10^{-3}	*
46	1.0×10^{-6}												8×10^{-7}
57	1.5×10^{-4}												*
2	$\ll 10^{-7}$				4×10^{-7}						1.0×10^{-5}		*
18	$\ll 10^{-7}$												*
		ΔP(F|E)/P(F|E) (range for all transients evaluated)											
All evaluated		−0.6 to −0.7	1 to 3	−0.01 to −0.2	1 to 4	−0.7 to −0.9	0.5 to 1.1	0.5 to 1.5		0.02 to 0.2	5 to 25	0.9 to 1	< 0.1 to 0.8

*Decrease in P(F|E) < 10%.

for the idealized transients, and the results of such a study are presented in Table 5.6.

Table 5.6 shows the time of WPS for each of the transients considered and the calculated values of $P(F|E)$ with and without WPS included in the analysis. It is apparent that for these idealized transients WPS has a very significant effect.

5.5.5 Effect of Proposed Remedial Measures on $P(F|E)$

The proposed remedial measures considered in the fracture-mechanics studies were (1) reduction in the fluence rate, (2) in-service inspection for flaws in the vessel, (3) a limit on repressurization, and (4) annealing of the vessel.

5.5.5.1 Reduction in Fluence Rate

The reduction in fluence rate was assumed to take place on January 1, 1985, and it was assumed to be the same at all critical locations in the vessel wall. The effect was simply to change the proportionality constant between F_o and EFPY beyond January 1, 1985. At this time the vessel will have been in service for ~7 EFPY, and the fluence for weld SA1430 will be 2.76×10^{18} neutrons/cm^2. The information in Figure 5.9 is based on a fluence rate for weld SA1430 of 3.26×10^{17} neutrons/(cm^2·EFPY) for the period 5 EFPY to 32 EFPY. The number of EFPY corresponding to the fluence in Figure 5.9 after January 1, 1985, is

$$\mathrm{EFPY} = 7 + 3.07 \times 10^{-18} f(F_o - 2.76 \times 10^{18}) \qquad (5.19)$$

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Table 5.6. Effect of including WPS in calculation of P(F|E) at 32 EFPY

Transient	P(F|E) w/o WPS	Time of WPS (min)	P(F|E) with WPS
1	3×10^{-7}	27	$<10^{-7}$
1A, 5, 7	5.2×10^{-6}	18	
4, 9	2.0×10^{-3}	22	$<10^{-6}$
6	1.7×10^{-6}	28	$<10^{-7}$
10	9.8×10^{-6}	28	$<10^{-7}$
14	1.3×10^{-3}	36	$<10^{-6}$
21	9×10^{-7}	40	$<10^{-7}$
24	1.0×10^{-4}	18	9×10^{-7}
25	5.2×10^{-4}	17	3×10^{-5}
26, 27, 32	6.2×10^{-4}	28	3×10^{-4}
28, 30	4.0×10^{-4}	18	4×10^{-5}
29	3.0×10^{-3}	20	$<2 \times 10^{-6}$
33	6.5×10^{-4}	18	3×10^{-6}
34	1.8×10^{-3}	18	5×10^{-5}
41	1.8×10^{-3}	30	$<10^{-6}$
44	5.4×10^{-3}	22	$<10^{-6}$
46	1.0×10^{-6}	20	$<10^{-7}$
57	1.5×10^{-4}	20	1×10^{-5}

where

f = factor by which fluence rate is reduced on January 1, 1985

F_o = inner-surface fluence for weld SA1430

For this particular study, fluence reduction factors of 2, 4, and 8 were examined. Using these factors in Eq. 5.19 results in F_o (32 EFPY) = 6.84×10^{18}, 4.80×10^{18}, and 3.78×10^{18} neutrons/cm^2, respectively, compared to 10.9×10^{18} neutrons/cm^2 for $f = 1$. The corresponding values of $P(F|E)$ for the transients included in Figure 5.9 can be estimated by entering these fluences on the abscissa of Figure 5.9. (The EFPY scale is, of course, incorrect except for $f = 1$.) These results are summarized in Table 5.7.

5.5.5.2 In-service Inspection

In-service inspection was assumed to have a capability of finding either 90% or 99% of the surface flaws with depths equal to or greater than 6 mm and none of the flaws with depths less than 6 mm. It was further assumed that all flaws found would be repaired. If before the in-service inspection, no calculated failures were attributed to initial flaws with depths less than 6 mm, then the 90% and 99% inspection would reduce $P(F|E)$ by factors of 0.1 and 0.01, respectively. However, in most cases the very shallow flaws do result in failure. The calculated effect of in-service inspection under these conditions is indicated in Table 5.7 for a time in the life of the vessel of 32 EFPY.

5.5.5.3 Limit on Repressurization

The effect on $P(F|E)$ of a limit on repressurization was investigated by specifying a single limit of 6.9 MPa. The benefit of this limit is

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Table 5.7. Benefit of remedial measures, $P(F|E)_{wRM}/P(F|E)_{w/oRM}$* at 32 EFPY

Transient	P(F|E) Original Mean Values at 32 EFPY	Reduction in Fluence Rate on Jan. 1, 1985			In-service Inspection		Limit on Repressurization	Annealing at 9 EFPY
		Rate Reduction Factor			Inspection Capability**			
		2	4	8	90%	99%		
1	3×10^{-7}	***	***	***				
1A, 5, 7	5.2×10^{-6}	0.08	***	***	0.13	0.04		0.19
2	$<<10^{-7}$	***	***	***				
4, 9	2.0×10^{-9}	0.4	0.2	0.03	0.29	0.22		0.50
6	1.7×10^{-6}	***	***	***				0.10
10	9.8×10^{-6}	0.2	***	***			0.03	0.30
14	1.3×10^{-3}	0.4	0.2	0.08	0.28	0.21	0.13	0.54
18	$<<10^{-7}$	***	***	***				
21	9×10^{-7}	***	***	***				
24	1.0×10^{-4}	0.2	0.05	0.02	0.15	0.06		0.37
25	5.2×10^{-4}	0.4	0.1	0.06	0.21	0.13		0.42
26, 27, 32	6.2×10^{-4}	0.2	0.05	0.02	0.55	0.51	***	0.35
28, 30	4.0×10^{-4}	0.2	0.08	0.03	0.50	0.45	***	0.38
29	3.0×10^{-3}	0.3	0.1	0.03	0.56	0.51	0.07	0.43
33	6.5×10^{-4}	0.3	0.1	0.05				0.46
34	1.8×10^{-3}	0.4	0.2	0.1				0.61
41	1.8×10^{-3}	0.3	0.1	0.05	0.31	0.24		0.56
44	5.4×10^{-3}	0.6	0.2	0.1	0.38	0.31	0.02	0.65
46	1.0×10^{-6}	***	***	***				0.28
57	1.5×10^{-4}	0.1	0.02	***			0.01	

*$P(F|E)_{wRM}$: conditional probability of failure with remedial measure. $P(F|E)_{w/oRM}$: conditional probaility of failure without remedial measure.

**Refer to text for an explanation of inspection capability and assumption regarding repair of flaws.

***$P(F|E) <10^{-7}$.

presented in Table 5.7 for 32 EFPY. For several of the transients the pressure does not drop below 6.9 MPa, and for these transients the limit on repressurization does not apply.

5.5.5.4 Annealing

Annealing of the pressure vessel will increase the fracture toughness of the vessel material, and the amount of the increase will depend on the annealing temperature and time, the chemistry of the material, and the number of times the vessel is annealed. Test results from small specimens indicate that essentially full recovery of the initial fracture toughness might be achieved by annealing in the temperature range 400-450°C for ~200 h.[14] Although preliminary studies indicate that such a process would probably be feasible in some PWR plants, the feasibility of annealing the Oconee-1 vessel under these conditions has not been established. Nevertheless, for the purpose of this study it was assumed that the Oconee-1 vessel would be annealed when the plant achieved ~9 EFPY (~January 1987). In effect, 9 yr would be subtracted from the EFPY scale in Figure 5.9 for EFPY ≥ 9. The benefit at 32 EFPY of this assumed annealing situation is indicated in Table 5.7.

REFERENCES

1. F. J. Loss, R. A. Gray, Jr., and J. R. Hawthorne, "Significance of Warm Prestress to Crack Initiation During Thermal Shock," Report NRL/NUREG-8165, Naval Research Laboratory, NTIS, September 29, 1977.

2. R. D. Cheverton and S. K. Iskander, "Thermal-Shock Investigations," in "Heavy-Section Steel Technology Program Quarterly Progress Report October-December 1980," NUREG/CR-1941 (ORNL/NUREG/TM-437), pp. 37-54, Union Carbide Corporation-Nuclear Division, Oak Ridge National Laboratory.

3. R. D. Cheverton and D. G. Ball, "OCA-P, A Probabilistic Fracture-Mechanics Code for Application to Pressure Vessels," NUREG/CR-3618 (ORNL-5991), Union Carbide Corporation-Nuclear Division, Oak Ridge National Laboratory (in preparation).

4. R. D. Cheverton, D. G. Ball, and S. E. Bolt, "Thermal-Shock Investigations," in "Heavy-Section Steel Technology Program Quarterly Progress Report April-June 1983," NUREG/CR-3334, Vol. 2 (ORNL/TM-8787/V2), pp. 57-74, Union Carbide Corporation-Nuclear Division, Oak Ridge National Laboratory.

5. T. U. Marston (Ed.), "Flaw Evaluation Procedures, ASME Section XI, Background and Application of ASME Section XI, Appendix A, Special Report," EPRI NP-719-SR, American Society of Mechanical Engineers, Electric Power Research Institute, August 1978.

6. Letter from F. J. Loss (NRL) to R. H. Bryan (ORNL), March 31, 1981.

7. NRC staff evaluation of Pressurized Thermal Shock, Sept. 13, 1982 draft, attachment to transmittal document, Secy 82-465, Nov. 23, 1982. (Available in NRC public document room.)

8. R. D. Cheverton and D. G. Ball, "Thermal-Shock Investigations," in "Heavy-Section Steel Technology Program Quarterly Progress Report October-December 1982," NUREG/CR-2751, Vol. 4 (ORNL/TM-8369/V4), p. 68, Union Carbide Corporation-Nuclear Division, Oak Ridge National Laboratory.

9. ASME Boiler and Pressure Vessel Code, Section III, Division I, Subsection NA, Appendix I, American Society of Mechanical Engineers, New York, 1974.

10. R. M. Gamble and J. Strosnider, Jr., "An Assessment of the Failure Rate for the Beltline Region of PWR Pressure Vessels During Normal Operation and Certain Transient Conditions," NUREG-0778, June 1981.

11. W. Marshall, "An Assessment of the Integrity of PWR Pressure Vessels," United Kingdom Atomic Energy Authority, Second Report, March 1982.

12. R. Y, Rubinstein, "Simulation and the Monte Carlo Method," Israel Institute of Technology, John Wiley & Sons, New York, 1981, pp. 115-117.

13. "Reactor Vessel Pressurized Thermal Shock Evaluation," DPC-RS-1001, Duke Power Company Oconee Nuclear Station, January 1982.

14. J. R. Hawthorne, Naval Research Laboratory, personal communication to R. D. Cheverton, Oak Ridge National Laboratory.

6.0 PTS INTEGRATED RISK AND POTENTIAL MITIGATION MEASURES

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6.1 Introduction

The preceding three chapters have outlined the procedures and techniques employed to estimate the three fundamental parameters (transient frequency, thermal-hydraulic history, and the conditional probability of vessel failure) required to quantify the risk inherent in a PTS transient. This chapter discusses the means by which these three influences are integrated to yield an estimated frequency of vessel failure (through-wall crack penetration), the results of this integration process, and, briefly, the impact of potential mitigating actions which could be applied to an Oconee type plant.

6.2 Risk Integration

In order to quantify the conditional probability of vessel failure for each endstate of the initiator-specific asymptotic event trees (i.e., Table A.2), two further estimations are required. First, the influence of operator actions in modifying the course of a particular asymptotic transient on the transient frequency must be quantitatively estimated, and second, a thermal-hydraulic history reflecting the operator actions must be assigned to each of the transients. The thermal-hydraulic history will directly relate a transient to fracture mechanics calculations of the vessel failure probability, given the occurrence of the transient.

While this approach is theoretically appealing, there are significant operational and resource problems associated with it. In the first place, the thermal-hydraulic histories of approximately 6000 transients (248 asymptotic transients x ~20 operator action combinations) would have to be calculated or estimated. Moreover, the conditional failure probability for each transient would have to be calculated using the fracture-mechanics techniques outlined previously. Such an effort was clearly beyond the resources available to this project, and an alternate, albeit less precise, procedure was employed.

As noted in Chapter 4.0, a limited number of event sequences were calculated in detail using the TRAC and RELAP codes. These sequences served as a basis for the estimation of approximately 30 additional sequences. Fracture mechanics calculations (Section 5.0) were then performed on a subset of the total of approximately 60 calculated and estimated thermal-hydraulic histories as a function of the fluence sustained by the vessel. The results of the FM calculations serve as the basis for estimating the required $P(TWC/E_i)$ for the transients. Table 6.1 summarizes the set of thermal-hydraulic/fracture mechanics data available for use in the risk integration procedure.

One consequence of the limited number of transients that were analyzed is a severe limitation on the number of operator actions that could be modeled. In most cases, the operator action trees described in Chapter 3.0 were collapsed to account only for the most significant action of those possible (based on engineering judgment). For most PTS transients, this corresponds to the isolation of affected steam generator. The failure branch is then defined as the lack of isolation on the part of the operator. In certain cases (specifically those involving a loss of all feedwater), the potential for operator errors of commission was incorporated. An example of this

Table 6.1 Oconee-1 PTS transient data
[P(F|E) vs fluence for weld SA1430]

DATA TABLE

TNAME	TRANS#	FLUENCE					
		@0.273	@0.545	@0.818	@1.09	@1.417	
NONE	0	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	
TBVG1	1	2.8E-10	6.9E-09	3.0E-08	4.0E-07	2.4E-06	RECALC
TBVG2	2	3.5E-10	8.5E-09	3.7E-08	1.0E-07	2.1E-07	MIN
TBVG3	3	3.5E-10	8.5E-09	3.7E-08	1.0E-07	2.1E-07	MIN
TBVG4	4	6.0E-05	4.0E-04	1.1E-03	2.0E-03	3.4E-03	
TBVG5	5	1.0E-08	2.6E-07	1.1E-06	5.2E-06	1.6E-05	
TBVG6	6	5.9E-09	1.5E-07	6.2E-07	1.7E-06	3.6E-06	
TBVG7	7	1.0E-08	2.6E-07	1.1E-06	5.2E-06	1.6E-05	
TBVG8	8	3.5E-10	8.5E-09	3.7E-08	1.0E-07	2.1E-07	MIN
TBVG9	9	6.0E-05	4.0E-04	1.1E-03	2.0E-03	3.4E-03	
TBVG10	10	3.5E-08	8.5E-07	3.7E-06	1.0E-05	2.1E-05	
PSBG1	11	3.5E-10	8.5E-09	3.7E-08	1.0E-07	2.1E-07	MIN
PSBG2	12	3.5E-10	8.5E-09	3.7E-08	1.0E-07	2.1E-07	MIN
PSBG3	13	3.5E-10	8.5E-09	3.7E-08	1.0E-07	2.1E-07	MIN
PSBG4	14	4.8E-05	2.8E-04	7.0E-04	1.3E-03	2.0E-03	
PSBG5	15	3.5E-10	8.5E-09	3.7E-08	1.0E-07	2.1E-07	MIN
OVRFD1	16	3.5E-10	8.5E-09	3.7E-08	1.0E-07	2.1E-07	MIN
OVRFD2	17	3.5E-10	8.5E-09	3.7E-08	1.0E-07	2.1E-07	MIN
OVRFD3	18	3.5E-10	8.5E-09	3.7E-08	1.0E-07	2.1E-07	MIN
OVRFD4	19	3.5E-10	8.5E-09	3.7E-08	1.0E-07	2.1E-07	MIN
OVRFD5	20	3.5E-10	8.5E-09	3.7E-08	1.0E-07	2.1E-07	MIN
OVRFD6	21	1.2E-09	2.8E-08	1.2E-07	1.4E-06	6.6E-06	
OVRFD7	22	3.5E-10	8.5E-09	3.7E-08	1.0E-07	2.1E-07	MIN
SGTR	23	3.5E-10	8.5E-09	3.7E-08	1.0E-07	2.1E-07	MIN
ZP1TBV	24	3.6E-07	8.8E-06	3.8E-05	1.0E-04	2.2E-04	
ZP2TBV	25	8.4E-06	7.8E-05	2.4E-04	5.2E-04	2.2E-04	
MSLB1	26	2.2E-06	5.0E-05	2.4E-04	6.2E-04	1.3E-03	
MSLB2	27	2.2E-06	5.0E-05	2.4E-04	6.2E-04	1.3E-03	
MSLB3	28	2.4E-06	4.2E-05	1.6E-04	4.0E-04	8.6E-04	
MSLB4	29	3.8E-05	4.2E-04	1.4E-03	3.0E-03	5.4E-03	
MSLB5	30	2.4E-06	4.2E-05	1.6E-04	4.0E-04	8.6E-04	
MSLB6	31	6.7E-09	6.3E-08	1.9E-07	4.2E-07	8.0E-07	TBVG1
MSLB7	32	2.2E-06	5.0E-05	2.4E-04	6.2E-04	1.3E-03	
W1TBV	33	1.2E-05	1.0E-04	3.2E-04	6.5E-04	1.2E-03	
W2TBV	34	8.3E-05	4.4E-04	1.1E-03	1.8E-03	2.9E-03	
LANL1	35	3.5E-10	8.5E-09	3.7E-08	1.0E-07	2.1E-07	MSLB(BASE)-MIN
LANL2	36						MSLB(PARA1)
LANL3	37						MSLB(PARA2)
LANL4	38						MSLB(PARA3)
LANL5	39	6.7E-09	6.3E-08	1.9E-07	4.2E-07	8.0E-07	PORV LOCA
LANL6	40						TURBINE TRIP
LANL7	41	4.2E-05	3.2E-04	9.0E+04	1.8E-03	3.0E-03	TBV(5A)
LANL8	42	3.5E-10	8.5E-09	3.7E-08	1.0E-07	2.1E-07	TBV(5B)-MIN
LANL9	43	3.5E-10	8.5E-09	3.7E-08	1.0E-07	2.1E-07	TBV(5C)-MIN
LANL10	44	4.0E-04	1.7E-03	3.6E-03	5.4E-03	8.0E-03	TBV(6A)
LANL11	45	3.5E-10	8.5E-09	3.7E-08	1.0E-07	2.1E-07	TBV(6B)-MIN
LANL12	46	8.7E-09	2.1E-07	9.1E-07	2.5E-06	5.3E-06	TBV(6C)
LANL13	47	3.5E-10	8.5E-09	3.7E-08	1.0E-07	2.1E-07	LOCA 2"-MIN
LANL14	48	3.5E-10	8.5E-09	3.7E-08	1.0E-07	2.1E-07	LOCA 4"-MIN
LANL15	49	6.7E-09	6.3E-08	1.9E-07	4.2E-07	8.0E-07	RAN.SECO
LANL16	50						CR
INEL1	51						MSLB
INEL2	52	6.7E-09	6.3E-08	1.9E-07	4.2E-07	8.0E-07	PORV LOCA
INEL3	53	3.5E-10	8.5E-09	3.7E-08	1.0E-07	2.1E-07	EFW OVERFEED-MIN
INEL4	54						TURBINE TRIP
INEL5	55	3.5E-10	8.5E-09	3.7E-08	1.0E-07	2.1E-07	LOCA 2"-MIN
INEL6	56	6.7E-09	6.3E-08	1.9E-07	4.2E-07	8.0E-07	MAX MFW
INEL7	57	2.5E-07	6.1E-06	4.5E-05	1.5E-04	4.1E-04	4TBVS@HS
INEL8	58	6.7E-09	6.3E-08	1.9E-07	4.2E-07	8.0E-07	PUMP SUCTION 2.5
INEL9	59	3.2E-08	3.0E-07	9.2E-07	2.0E-06	3.8E-06	SGTR-MIN
MIN	98	3.5E-10	8.5E-09	3.7E-08	1.0E-07	2.1E-07	
DEF	99	4.0E-04	1.7E-03	3.6E-03	5.4E-03	8.0E-03	

type of error is the restoration of EFW in an uncontrolled manner. It must be noted, however, that certain operator actions, in particular the trip/no trip of the RCPs on a valid ECCS demand, were neglected.

By examining the system state categories for each transient listed in Table A.2 together with the transient initiator, the most significant operator action(s) were determined, and the probability of each end-state on the collapsed operator action yields an estimate of a transient frequency that accounts for operator intervention. Then, from the set of available transients, the corresponding thermal-hydraulic history was assigned to each of the transients. For those transients for which a detailed event description was unavailable (i.e., those transients whose individual asymptotic frequencies were less than 1.0E-06), a default transient was assigned. This default transient represented the highest conditional probability of failure obtained of all the transients calculated. The default transient is LANL10 in Table 6.1, the LANL calculation of 4 TBVs failed open after reactor trip from full power (see Chapter 4.0, Section 4.3.2.4 for a description of this sequence). Table A.3 summarizes the results of the transient assignment procedure for the 498 transients obtained by incorporating operator actions into the 295 transients listed in Table A.2. Also listed in Table A.3 are the conditional probability of failure, given the occurrence of the transient $P(F|E)$ (at a fluence equivalent to 32 EFPY), and the contribution to the overall frequency of vessel failure $\phi(TWC)$. By summing overall transients in Table A.3, an estimated frequency of vessel failure, for the initiators considered, of 4.5 E-06/RY is obtained.

In order to summarize the results, each transient listed in Table A.3 was also classified by type of malfunction involved. For example, a FW (feedwater) transient incorporates some malfunction of the feedwater system, an SL (steam line) transient involves a pressure control malfunction, etc. Also, a combined transient type (SS for secondary side) was used to describe those transients that involve both feedwater and steam line problems. Table 6.2 summarizes the frequency of vessel failure (as a function of fluence) for each transient type. As indicated, the dominant transient type is labeled RES (i.e., residual) and accounts for those transients whose individual frequencies were less than 1.0E-06 but whose cumulative frequencies are significant. The total frequency of vessel failure and the individual contributions of the various identified initiators as a function of EFPY fluence, and RTNDT are depicted graphically on Figures 6.1a, 6.1b, and 6.1c.

Finally, it is instructive to examine those transients that dominate the estimated frequency of vessel failure. Table 6.3 lists those transients whose contribution to the overall frequency of vessel failure are >1.0E-08. These 21 transients account for approximately 95% of the overall frequency and, as can be seen by inspection, are dominated by the residual components (59% of the total), particularly those originating from the reactor trip initiator (21% of the total).

Table 6.2. Breakdown of ϕ(TWC) by transient type

		Fluence for Weld SA1340 (10^{19} neutrons/cm^2)				
TYPE	TFREQ	0.273	0.595	0.818	1.09	1.417
FW	2.6-01	4.3-10	4.1-09	1.4-08	3.4-08	6.7-08
SL	9.8-01	1.8-08	1.8-07	6.7-07	1.6-06	3.1-06
SS	2.7-02	2.9-09	2.1-08	5.8-08	1.1-07	2.0-07
LOCA	1.6-01	6.1-10	4.3-09	1.3-08	3.0-08	5.8-08
SGTR	9.9-03	1.3-10	8.1-10	2.2-09	4.3-09	7.3-09
RES	5.0-04	2.0-07	8.5-07	1.8-06	2.7-06	4.0-06
TOTAL		2.2-07	1.1-06	2.6-06	4.5-06	7.5-06

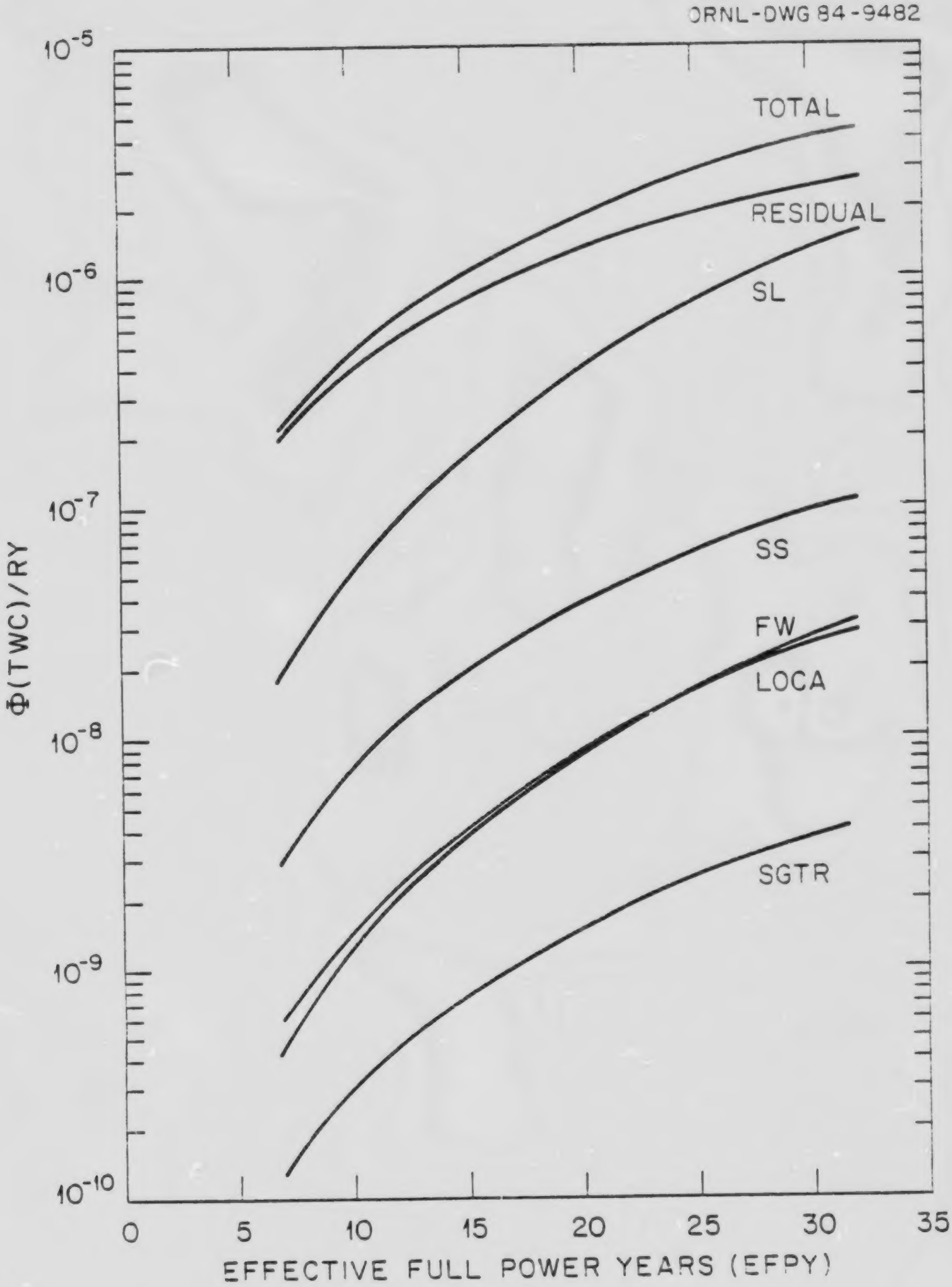

Figure 6.1a Total and component frequencies of vessel through wall crack as a function of effective full power years.

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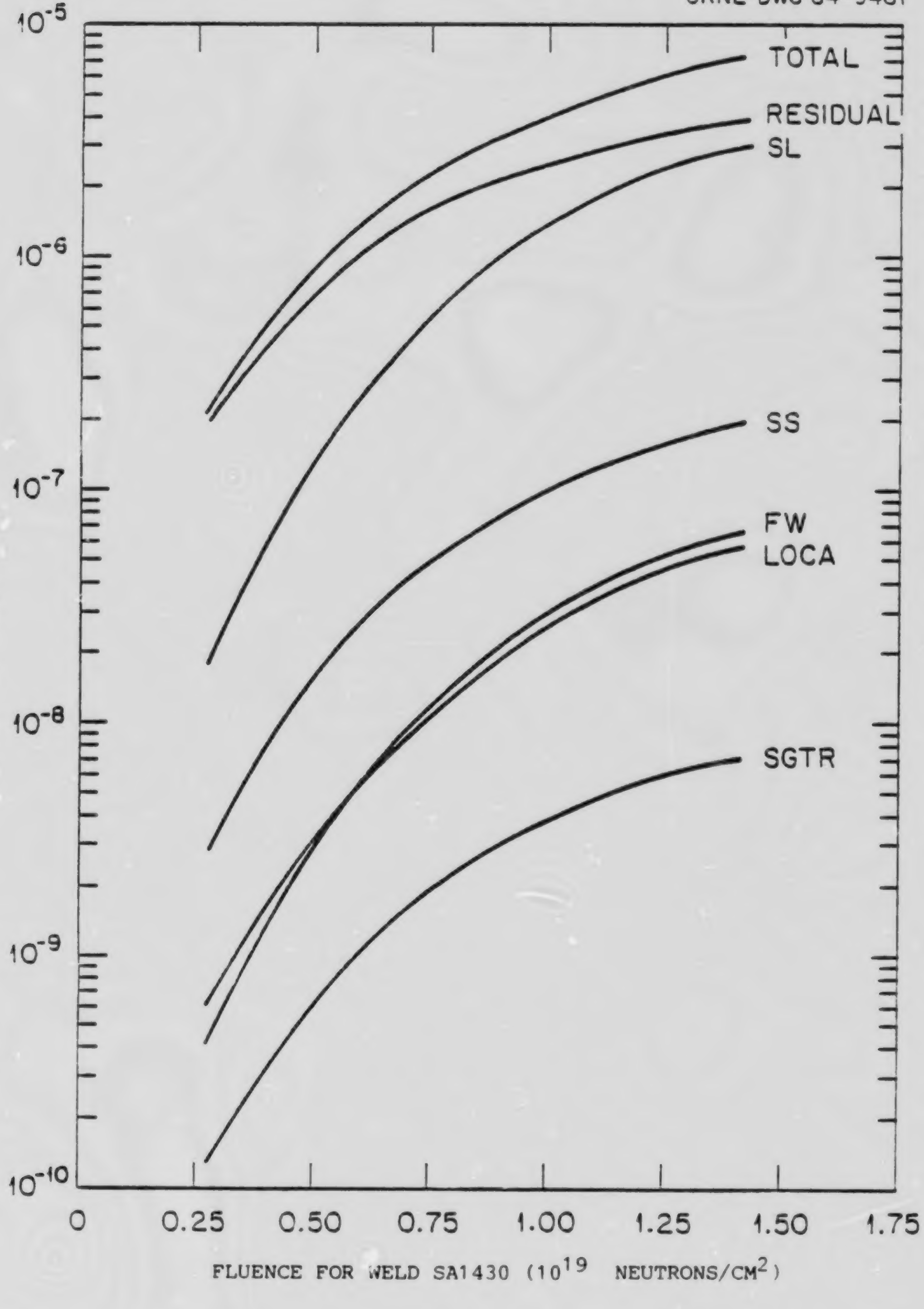

Figure 6.1b Total and component frequencies of vessel through wall crack as a function of fluence.

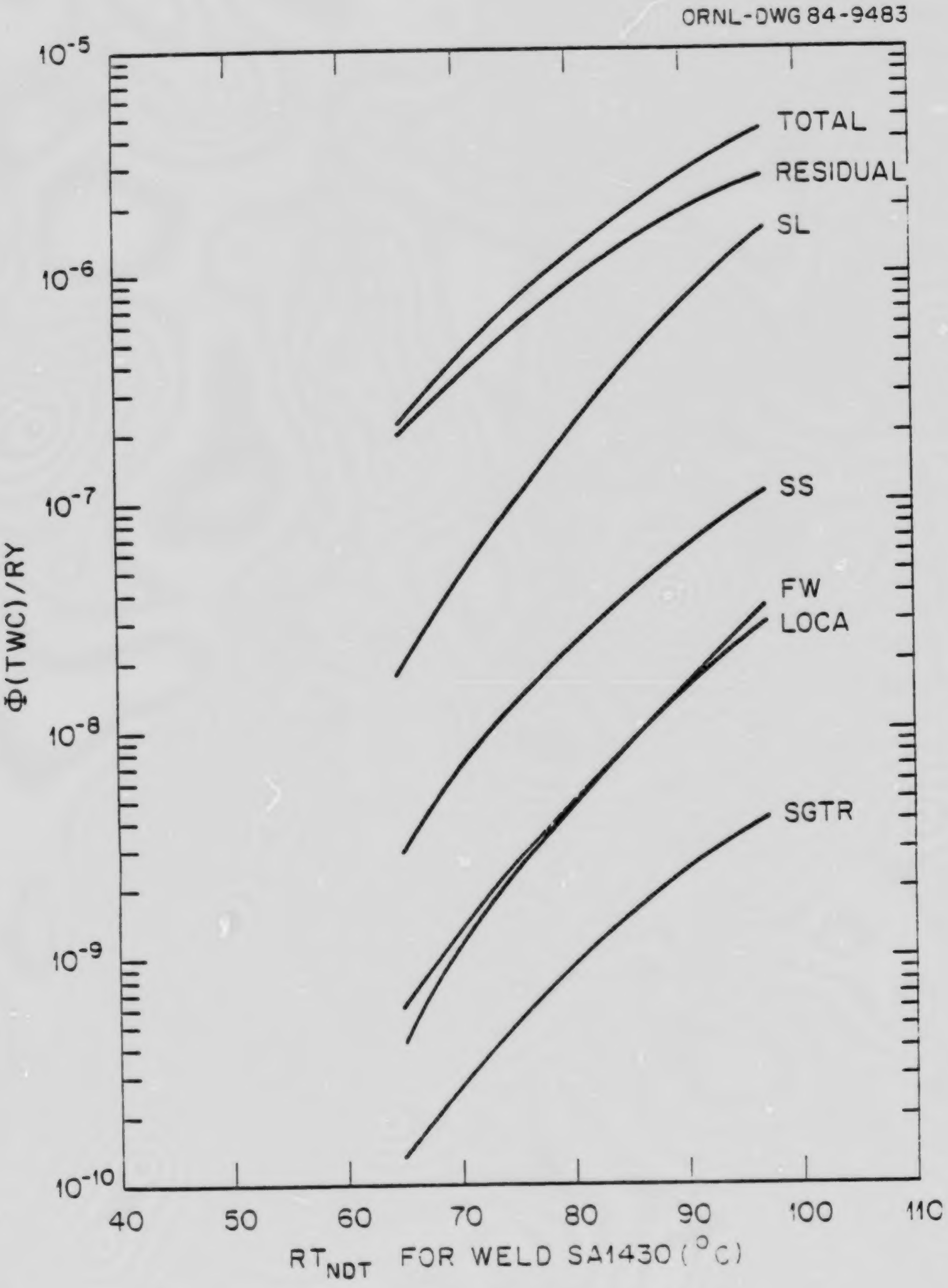

Figure 6.1c Total and component frequencies of vessel through wall crack as a function of RT_{NDT}.

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Table 6.3 Dominant sequences contributing to Oconee-1 PTS risk

REC	TYPE	INIT	SPC	FWS	ECC	PPC	EFREQ	TRANS	HEP	TRANS#	TFREQ	P(F/T)	P(TWC)
3	FW	RT	0	1	0	0	1.6E-01	OVRF03	9.9E-01	18	1.6E-01	1.0E-07	1.6E-08
24	SL	RT	1	0	0	0	6.9E-01	TBVG2	9.9E-01	2	6.8E-01	1.0E-07	6.8E-08
53	SL	RT	1	1	0	0	2.2E-02	TBVG4	1.0E-02	4	2.2E-04	2.0E-03	4.4E-07
116	SS	RT	3	1	0	0	2.4E-03	TBVG9	1.0E-02	9	2.4E-05	2.0E-03	4.8E-08
147	SL	RT	6	0	0	0	5.0E-04	MSLB1	9.9E-01	26	5.0E-04	6.2E-04	3.1E-07
148	SL	RT	6	0	0	0	5.0E-04	TBVG9	1.0E-02	9	5.0E-06	2.0E-03	1.0E-08
151	SL	RT	6	0	0	2	3.1E-05	MSLB7	9.9E-01	32	3.1E-05	6.2E-04	1.9E-08
158	RES	RT	99	99	99	99	1.9E-04	DEF	1.0E+00	99	1.9E-04	5.4E-03	1.0E-06
208	SS	EMFW	3	1	0	0	1.3E-03	TBVG9	1.0E-02	9	1.3E-05	2.0E-03	2.6E-08
228	RES	EMFW	99	99	99	99	7.2E-05	DEF	1.0E+00	99	7.2E-05	5.4E-03	3.9E-07
229	SL	LSLB	0	0	0	0	9.6E-04	MSLB1	9.9E-01	26	9.5E-04	6.2E-04	5.9E-07
230	SL	LSLB	0	0	0	0	9.6E-04	TBVG9	1.0E-02	9	9.6E-06	2.0E-03	1.9E-08
231	SL	LSLB	1	1	0	0	3.0E-05	MSLB3	9.9E-01	28	3.0E-05	4.05-04	1.2E-08
235	RES	LSLB	99	99	99	99	5.3E-06	DEF	1.0E+00	99	5.3E-06	5.4E-03	2.9E-08
255	RES	SSLB	99	99	99	99	1.4E-05	DEF	1.0E+00	99	1.4E-05	5.4E-03	7.6E-08
343	SL	LOMFW	6	0	0	0	6.1E-05	MSLB1	9.9E-01	26	6.0E-05	6.2E-04	3.7E-08
344	SL	LOMFW	6	0	0	0	6.1E-05	MSLB1	9.9E-01	26	6.0E-05	6.2E-04	3.7E-08
350	RES	LOMFW	99	99	99	99	5.9E-05	DEF	1.0E+00	99	5.9E-05	5.4E-03	3.2E-07
418	RES	SBL1	99	99	99	99	7.8E-05	DEF	1.0E+00	99	7.8E-05	5.4E-03	4.2E-07
442	RES	SBL2	99	99	99	99	1.8E-05	DEF	1.0E+00	99	1.8E-05	5.4E-03	9.9E-08
467	RES	SGTR	99	99	99	99	1.8E-05	DEF	1.0E+00	99	1.8E-05	5.4E-03	9.9E-08
500	RES	SI	99	99	99	99	3.9E-05	DEF	1.0E+00	99	3.9E-05	5.4E-03	2.1E-07

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6.3 Potential Mitigation Measures

The effects of potential mitigating actions were examined as a part of this study. This section is not intended as a list of recommendations but is provided to give information on the relative value of actions which could be taken provided a need to reduce the integrated risk of a through-the-wall crack due to PTS is identified. In addition, it should be noted that the safety impact of mitigating actions on other types of events has not been performed and the cost benefit of these actions has not been evaluated at this time.

6.3.1 Limit Primary System Repressurization

The vessel failure probability is very sensitive to primary pressure during the portion of an overcooling transient in which the vessel becomes vulnerable to brittle fracture. The action of the high head HPI system is to rapidly repressurize the primary system. If left unthrottled, the system may repressurize to the PORV set point pressure in less than 20 min. However, plant procedures call for throttling of HPI so that the plant P-T specification is not violated.

A potential mitigating measure for reducing PTS risk would be to assure that the primary repressurization be limited to 1000 psia. The effect of such a limit on repressurization was evaluated and it was found that the frequency of vessel through-the-wall cracking was reduced to 3.2×10^{-7}/RY from the current estimate of 4.5×10^{-6}/RY for 32 EFPY.

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6.3.2 Effect of High SG Level Trip System

As indicated in the system description, the SGs at the Oconee-1 plant are equipped with instrumentation designed to trip the MFW pumps, should the SG level reach 90%. Although the frequency of vessel failure of 4.5 x 10^{-6}/RY includes the influence of this system, it is desirable to assess the potential mitigative value of this system for plants which do not incorporate a high SG level protection system.

Rather than reanalyze all the sequences for the case of no-SG protection system, the effect of this system was roughly estimated. The principal effect of the proper operation of the system is to terminate an MFW overfeed transient. In the terminology of the event tree chapter, this switches a potential sustained MFW overfeed transient (FW category 2) to an unsustained MFW overfeed transient. Thus, one effect of the elimination of the SG protection system is that all FW category 1 transients become FW category 2 transients. Additionally, the elimination of the system reduces the frequency of EFW overfeeds because the principal means by which the EFW system is initiated (i.e., trip of the MFW pumps) is no longer possible.

In estimating the value of the SG protection system, only the change in the frequencies of FW categories 1 and 2 are considered. The slight offsetting change due to the reduction of EFW overfeeds is neglected, changing all previous FW category 1 transients to FW category 2 transients. Together with the appropriate, but more severe transient descriptions, this results in an overall frequency of vessel failure of 6.1 x 10^{-6}/RY. This represents a 35% increase in the overall failure frequency and is indicative of the importance of this particular system. It should be noted, however, that this is only a rough estimate of the value of the system.

A detailed reanalysis of the set of transients would be required for a more precise value. Additionally, it should be noted that the physical limitations of the system were essentially ignored in arriving at this estimate. In particular, the potential for self-mitigation of an MFW overfeed (i.e., the MFW pump turbines tripping because of low steam quality caused by the overfeed) has been ignored.

6.3.3 Neutron Fluence Rate Reduction

The benefit from reducing the neutron fluence rate in the vessel wall by factors of 2, 4, and 8 were evaluated. Since fluence wields a cumulative impact on vessel RTNDT, the reduction in fluence rate will retard the effective rate of aging. Fluence-rate reduction factors of 2, 4, and 8 would result in PTS risk reductions by factors of approximately 2, 6, and 14 respectively at 32 EFPY.

6.3.4 Effect of HPI Heating

The high-pressure injection system is of concern in PTS evaluation because it introduces cold (50°F) liquid into the cold legs of the RCS primary close to the vessel downcomer. Various localized phenomena such as plume formation, metal-water heat transfer, fluid mixing in the cold leg, and fluid mixing in the downcomer region will determine whether the vessel wall will be subjected to temperatures significantly lower than the coolant bulk average temperature. As discussed in Chapter 4.0, Section 4.4, thermal-hydraulic analyses by Theophanous have determined that plume formation is not important for the LANL- and INEL-calculated cases where both high primary pressure and loop flow stagnation occur. Therefore, bulk mixing may

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be reasonably assumed for situations similar to those evaluated by Theophanous.

When injected in large quantity, HPI flow can cause significant cooling. However, the concept of heating the HPI to 90°F or 100°F will not greatly influence the temperature response of the Oconee-1 plant to overcooling events. The amount of system cooling due to HPI is a strong function of system temperature. This is illustrated by Figure 6.2. Simply stated, the greater the temperature difference between the HPI fluid and its final maximum temperature (say at the core exit), the greater the heat absorbed. As the maximum system temperature declines toward the HPI temperature, the cooling rate will approach zero. This behavior is expressed in Figure 6.2. Heating HPI to 90°F causes an offset of 2.9 MW (at HPI flow corresponding to operation at PORV set point pressure). Assuming a balance between heat input and cooling, a final temperature increase of close to 50°F could be realized. If another temperature-dependent cooldown mechanism, such as steam generator overfeeds or blowdown, is occurring, the temperature benefit of HPI heating will be reduced. For example, a transient with one TBV stuck open with failures to isolate feedwater and to throttle HPI will yield less than a 10°F benefit from raising HPI temperature to 90°F. Larger secondary-side cooldown mechanisms would further reduce the benefit; increases in HPI flow relative to the other cooldown mechanisms would increase the benefit.

The other aspect of HPI heating is the effect on localized cooling phenomena. Loop flow and vent valve flow combine with HPI flow to yield a weighted average temperature for the mixture. The relationship between

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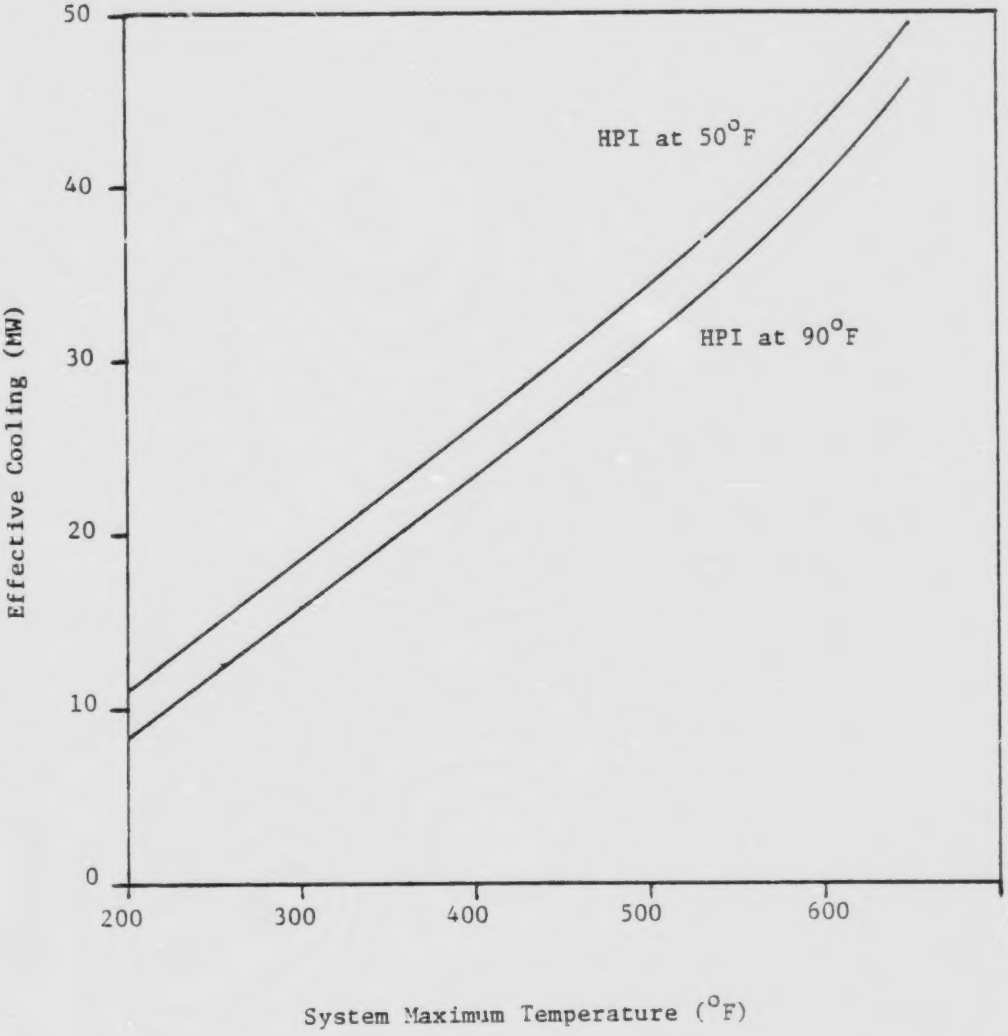

Figure 6.2 Primary cooling due to HPI flow at full repressurization (70 lb/s flow, 2450 psia).

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mixture temperature and flow ratio of loop and ventvalve flow to HPI flow is depicted in Figure 6.3. The figure contains sets of curves for mixing of 550°F loop flow with either 50°F or 90°F HPI and 300°F loop flow with 50°F or 90°F HPI flow. The loop temperature dominates the average temperature for flow ratios greater than 10 (loop):1 (HPI). HPI temperature dominates for ratios less than 0.1. At a flow ratio of 1.0, the benefit of HPI heating is one-half the HPI temperature difference, or 20°F. The benefit decreases for higher flow ratios and increases to the full 40°F value only for very low flow ratios. Also indicated on Figure 6.3 are the calculated flow ratios for several of the TRAC calculations performed by LANL marked loops A and B respectively. The solid markers indicate ratios based on loop flows alone. The dashed markers include both stagnated loop and vent valve flow in the calculations of flow ratio. For some cases the observed range is also noted. These cases suggest that the flowing loop in a transient will have a flow ratio on the order of 10 while a stagnated loop will have a ratio of about 1.0. Such conditions minimize the benefits to be gained from heating the HPI. Measures to restore loop flow would be of greater benefit in preventing localized cooldown situations.

Overall, HPI heating could provide some small but variable benefit in terms of downcomer temperature response. The sensitivity of the fracture mechanics calculation to low downcomer temperature may translate the very small temperature benefit into a measurable reduction of risk. However, other methods with greater potential for temperature improvement (such as loop flow augmentation, etc.) should be investigated first.

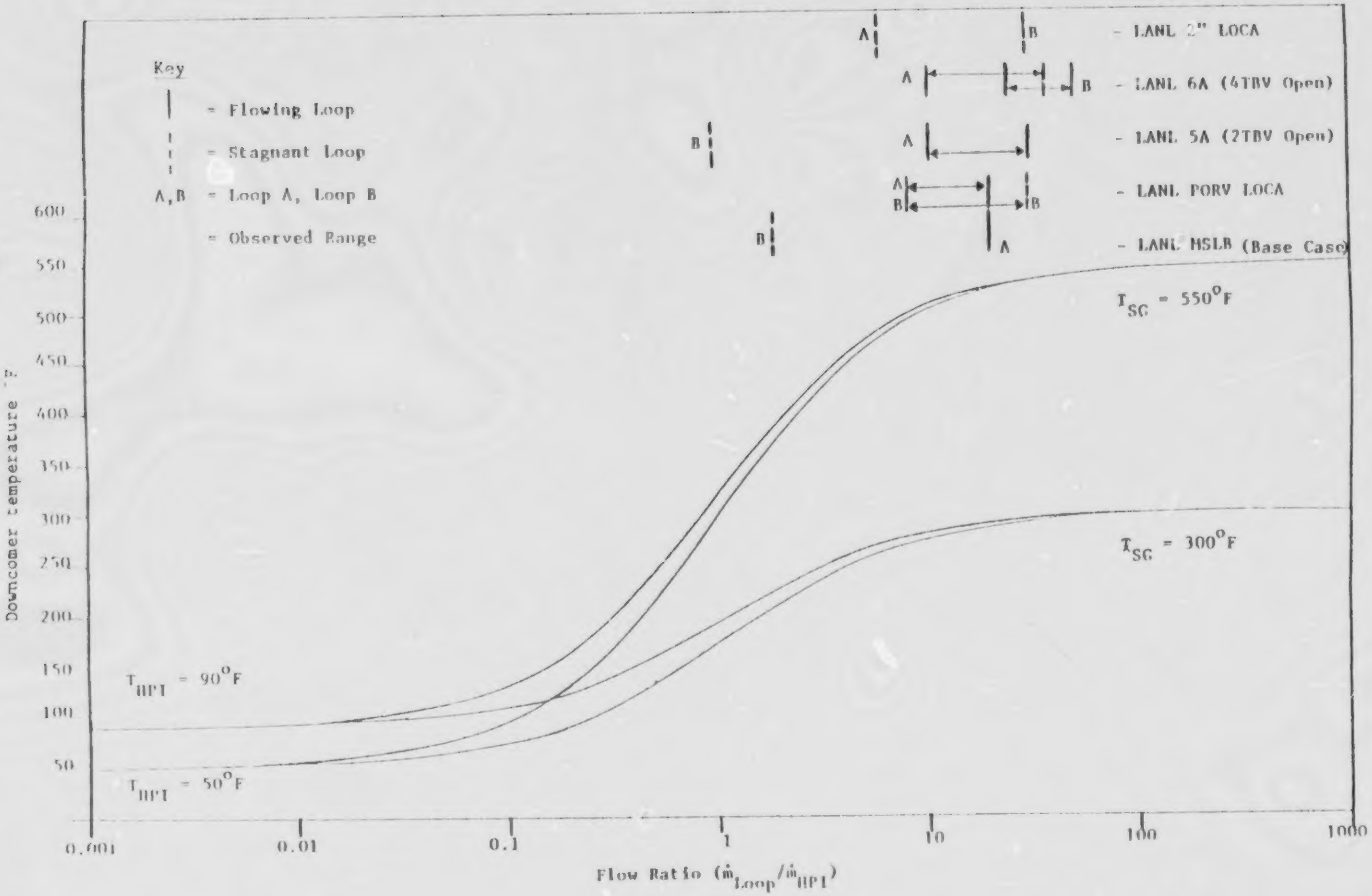

Figure 6.3 Downcomer temperature vs loop/HPI flow ratio and comparison with TRAC results.

6.3.5 In-Service Inspection

Improved in-service inspection techniques having the capability of detecting either 90% or 99% of the surface flaws with depths equal to or greater than 6 mm were evaluated. It was further assumed that all flaws found would be repaired. If, before the in-service inspection, no calculated failures were attributed to initial flaws with depths less than 6 mm, then the 90% and 99% inspection would reduce P(F/E) by factors of 0.1 and 0.01 respectively. However, in most cases the very shallow flaws which would not be detected or repaired make a significant contribution to failure. The calculated effect of in-service inspection under these conditions is indicated in Table 5.7 for an effective vessel age of 32 EFPY. For many severe transients, the detection and repair of 90% or 99% of all large flaws would reduce the probability of vessel failure by one-half. The reduction factor was limited by the fact that the very shallow flaws which would not be detected or repaired actually make a significant contribution to the total probability of vessel failure.

6.3.6 Vessel Annealing

Annealing of the vessel is presumed to restore the fracture thoughness of the vessel material to its pre-irradiated value effectively cancelling the effects of neutron fluence. The extent of recovery will depend on the chemistry of the vessel material, the time-temperature characteristics of the annealing procedure, and the number of times the vessel is annealed. Assuming full recovery, risk reduction by 1 to 2 orders of magnitude relative to the risk at 32 EFPY may be achieved. However, in some instances annealing may be required on some periodic basis if this measure is to prevent regrowth of the risk. The feasibility of in-place vessel annealing has not been addressed in sufficient detail to assure the effectiveness and practicality of this measure.

6.3.7 Operator Training

Operator training was not addressed specifically in this study. However, the benefits of very early action to isolate a broken steam generator or throttle HPI to limit repressurization of the primary were recognized. Written procedures direct the operators to make such beneficial action where applicable and could significantly mitigate many transients. The time required for the operators to analyze the situation and apply the specified corrective actions will therefore determine to a great extent the severity of a transient.

In the uncertainty analysis (Chapter 7), the sensitivity of the risk to operator action in the period of 10 to 20 min into a transient was observed to be small. For severe transients, actions taken in this period were too late to mitigate the accidents. For mild transients, the operator actions made little difference in what were already low PTS risk transients.

Thus, operator training may be of significant benefit if the response time to a severe overcooling transient can be reduced to under 10 minutes. Operator training may also be beneficial in reducing error rates below those assumed in this analysis and thus result in lower frequencies for some of the more severe transients.

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7.0 SENSITIVITY AND UNCERTAINTY ANALYSES

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7.1 Introduction

In addition to the estimate of the expected frequency of vessel failure, a complementary (and perhaps even more significant) analysis of the degree of uncertainty associated with the estimate is required. In fact, given some of the parameter estimation procedures used in this study, the expected frequency of vessel failure derived previously can only be regarded as a crude estimate. The purpose of this chapter is to assess the degree of confidence that can be associated with the estimate. This is done by considering most of the parameters used in this study as random variables and by estimating the influence of parameter variation on the final result, the probability of a through-wall vessel failure per reactor-year.

7.2 Uncertainty Analysis

In order to assess the uncertainty in risk of vessel failure, the frequency of failure is expanded about its estimated value using a Taylor series. Vessel failure is a function of event frequency, thermal-hydraulic conditions, and metallurgical state of the wall. The frequency of failure is expanded in terms of its independent variables as

$$\phi(TWC) - \hat{\phi}(TWC) = \frac{\delta\phi(TWC)}{\delta M}(M-\hat{M}) + \sum_i \frac{\delta\phi(TWC)}{\delta T(E_i)}[T(E_i) - \hat{T}(E_i)]$$
$$+ \sum_i \frac{\delta\phi(TWC)}{\delta\phi(E_i)}[\phi(E_i) - \hat{\phi}(E_i)] \qquad (7.1)$$
$$+ \text{ higher order terms}$$

where $\phi(TWC)$ = frequency of vessel failure (through-wall crack)

M = metallurgical parameters representing the state of the pressure vessel at the time event E_i occurs

$T(E_i)$ = thermal-hydraulic conditions (pressure, temperature, heat transfer coefficient) used to describe event E_i

$\phi(E_i)$ = frequency of occurrence of event E_i

The expression in Eq. 7.1 is used in finding the variance of the estimated frequency of failure, $\sigma^2[\phi(TWC)]$. The variance is the expected value of $[\phi(TWC) - \hat{\phi}(TWC)]^2$. Squaring the right-hand side of Eq. 7.1 and expanding,

$$\sigma^2[\phi(TWC)] = S_m^2\sigma_m^2 + \sum_i S_m S_{T(E_i)} \operatorname{cov}[M,T(E_i)] + \sum_i S_m S_{\phi(E_i)} \operatorname{cov}[M,\phi(E_i)]$$
$$+ \sum_i S_m S_{T(E_i)} \operatorname{cov}[M,T(E_i)] + \sum_i\sum_j S_{T(E_i)} S_{T(E_j)} \operatorname{cov}[T(E_i), T(E_j)]$$
$$+ \sum_i\sum_j S_{T(E_i)} S_{\phi(E_j)} \operatorname{cov}[T(E_i), \phi(E_j)]$$
$$+ \sum_i S_m S_{\phi(E_i)} \operatorname{cov}[M,\phi(E_i)] + \sum_i\sum_j S_{T(E_j)} S_{\phi(E_i)} \operatorname{cov}[T(E_j), \phi(E_i)]$$
$$+ \sum_i\sum_j S_{\phi(E_i)} S_{\phi(E_j)} \operatorname{cov}[\phi(E_i), \phi(E_j)]$$
$$+ \text{ higher order terms} \qquad (7.2)$$

Where S_x denotes the partial derivative with respect to the variable x, the variances are defined by

$$\sigma_x^2 = [x-\hat{x}]^2 p(x)dx \tag{7.3}$$

and the covariances by

$$\mathrm{cov}[x,y] = [y-\hat{y}][x-\hat{x}]p(x,y)dxdy \tag{7.4}$$

If it is assumed that

(1) M and $T(E_i)$ are independent variables,

(2) M and $\phi(E_i)$ are independent variables,

(3) $T(E_i)$ and $T(E_j)$ are independent for $i \neq j$,

(4) $T(E_i)$ and $\phi(E_j)$ are independent variables,

(5) $\phi(E_i)$ and $\phi(E_j)$ are independent for $i \neq j$,

and it is further assumed that higher order terms can be ignored, the variance of the result is given by

$$\sigma^2[\phi(TWC)] = S_m^2\sigma_m^2 + \sum_i S^2_{T(E_i)}\,\sigma^2_{T(E_i)} + \sum_i S^2_{\phi E_i)}\,\sigma^2_{\phi(E_i)} \tag{7.5}$$

Each of the components M, $T(E_i)$, and $\phi(E_i)$ of Eq. 7.5 actually represent a set of parameters, and each is considered separately when performing the calculations. First, a discussion of the assumptions necessary to get to the form of Eq. 7.5 is required.

7.3 Discussion of Assumptions

Most of the assumptions required to get to the form of Eq. 7.5 are reasonably accurate. For instance, it is likely that the metallurgical condition of the vessel wall at the beginning of a transient will be

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independent of the transient. Other assumptions are not as accurate, however. For instance, an implied assumption in the development has been that the errors associated with each variable are unbiased. This is obviously not the case.

One source of bias is the error associated with thermal-hydraulic modeling. When uncertainties are present, the tendency is to err on the side of conservatism. The many conservative assumptions present in the computer models tend to bias the pressures, temperatures, and heat transfer coefficients resulting from transient calculations.

The same type of bias is present in the metallurgical calculations. Where realistic numbers cannot be precisely obtained, conservative assumptions are made.

A third, and possibly the most significant, source of bias in the errors comes from the lumping of a large class of transients. For a given initiator many different events are possible, depending on the operator action. Of these many events, only two (or sometimes three) are analyzed, and the probability associated with all the events is distributed to those. The events examined are the most "significant" in terms of pressurized thermal shock. This tends to bias the results toward these significant events.

Another required assumption is that the covariance between all variables (thermal-hydraulic conditions, metallurgical parameters, and event frequencies) be zero. As mentioned, this is probably accurate for most variables. For some, however, the covariance may be significant.

This significant covariance, together with the bias discussed above, is enough to make the results of calculations using Eq. 7.5 little better than order-of-magnitude determination. Once this handicap is realized, the calculations may proceed as described in the following sections.

7.4 Transient Frequency Uncertainties

The frequency of a particular transient, E_i, depends on the initiator frequencies, the branch point probabilities, and the human error probabilities used in the operator action trees. These comprise the independent variables denoted by X_j. The conditional frequency of a particular transient resulting from a specific initiator, I_j, can be expressed as

$$\phi(E_i/I_j) = \phi(I_j) \prod_{k\varepsilon E_i} P\,(BP_R) \prod_{k\varepsilon E_i} P\,(HF_k) \tag{7.6}$$

The required partial derivatives are then given by

$$\frac{\delta\phi(TWC)}{\delta\phi(I_j)} = \sum_i P(F/M,T(E_i) \prod_{k\varepsilon E_i} P(BP_k) \prod_{k\varepsilon E_i} P\,(HF_k)$$

$$= P(F/I_j) \tag{7.7}$$

$$\frac{\delta\phi(TWC)}{\delta\phi P(PB_1)} = \sum_j \phi(I_j) \sum_i P(F/M,\ T(E_i) \prod_{\substack{k\varepsilon E_i \\ k\neq 1}} P(BP_k) \prod_{k\varepsilon E_i} P(HF_k)$$

$$= P(F/BP_1),\ \text{and} \tag{7.8}$$

$$\frac{\delta\phi(TWC)}{\delta\phi P(HF_1)} = \sum_j \phi(I_j) \sum_i P(F/M,\ T(E_i) \prod_{k\varepsilon E_i} P(BP_k) \prod_{\substack{k\varepsilon E_i \\ k\neq 1}} P(HF_k)$$

$$= P(F/HF_1) \tag{7.9}$$

As noted in Eq. 7.7 through 7.9, the partial derivatives required in the uncertainty analysis correspond to specific conditional probabilities, a

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fact that facilitates both their calculation and their interpretation. For example, Eq. 7.7 indicates that the required constant for each initiator examined is just the conditional probability of vessel failure, given the occurrence of the initiator. Similarly, the partial derivative required for the branch point contribution to the variance is just the conditional probability of failure given that branch point BP_k is present in the description of the transient E_i for each initiator summed over all initiators and E_i.

Table 7.1 contains the initiator frequencies and multiplicative uncertainty factors used in this study. These include the variance in the estimate of each initiator frequency and a partial derivative relating this variance to the variance in ϕ(TWC).

7.5 Branch Point Uncertainties

Due to the transient categorization process employed in estimating the thermal-hydraulic and fracture mechanic parameters, the analysis of the effect of branch point probability uncertainties on the frequency of vessel failure requires a two-step procedure. The branch point coefficients are estimated by first determining the conditional probability of vessel failure given occurrence of a certain category transient. Then the conditional probability of that category transient given a specific branch probability is calculated, i.e.,

$$P[\phi(E_i)\ BP_j] = \sum_k P[\phi(E_i)/Cat_k]P[Cat_k/BP_j] \quad (7.10)$$

A supplemental benefit of this two-stage process is that the conditional probabilities of vessel failure, given specific transient categories, are

Table 7.1 Initiator frequency analysis

Paramter	Frequency	Parital	Mu	Variance	Contribution
RT	6.0E+00	3.4E-07	6.2E+00	2.1E+00	2.4E-13
EMFW	1.0E-01	4.4E-06	1.3E-01	1.4E-02	2.3E-13
LSLB	1.0E-03	6.6E-04	1.2E-03	4.4E-07	1.9E-13
SSLB	1.0E-02	7.8E-06	1.3E-02	1.4E-04	8.7E-15
LOMFW	5.0E-01	8.5E-07	6.7E-01	3.6E-01	2.6E-13
SBL1	1.0E-01	8.4E-06	1.3E-01	1.4E-02	1.0E-12
SBL2	1.0E-02	1.1E-05	1.3E-02	1.4E-04	1.7E-14
SGTR	1.0E-02	1.0E-05	1.3E-02	1.4E-04	1.6E-14
SI	1.0E-02	2.1E-05	1.3E-02	1.4E-04	6.5E-14

obtained and can be used to estimate the PTS challenge on a system-functional basis.

Once the conditional probabilities of vessel failure for specific categories have been obtained, it is relatively straightforward to sum those end-states (E_i) that are classified as being in a specific category and that involve the branch point in question. The conditional probabilities of a specific category being selected can then be determined. Incorporating these values into Eq. 7.10 and using Eq. 7.7 and 7.9 permits the required partial derivatives to be determined.

Table 7.2 summarizes the results of the branch point sensitivity/uncertainty analysis for those end-states that meet the initial criterion of $\phi(E_i) >$ 1.0E-06. The analysis for the uncertainty attributable to those end-states not specifically identified (i.e., the residual uncertainty) is described below.

7.6 Residual Branch Point Analysis

Since the transients with estimated frequencies below 1.0E-6 were not examined, identification of a specific set of branch points comprising each transient is not possible. However, it is known that all the residual transients involve one branch point for each system state event. Thus, as an upper bound on uncertainty for the residual transients, it is assumed that the branch point having the largest uncertainty for a given system state event was selected. Transients defined by this selection of branch points do not necessarily correspond to either the specific transients obtainable using the system state trees or even physically possible

Table 7.2 Branch point analysis summary

Type	Mu	Variance	Partial	Contribution	Branch:	#
SPC	9.5E-04	7.2E-07	3.5E-07	8.8E-20	TSV5-1	1
SPC	1.0E-03	8.0E-07	5.3E-06	2.3E-17	TBV5-01	1
SPC	1.0E-04	8.0E-09	1.2E-06	1.1E-20	TBV5-01	2
SPC	8.0E-02	5.1E-03	3.8E-06	7.3E-14	SSRV-01	1
SPC	1.0E-01	6.0E-03	2.2E-06	3.9E-14	TSVS-2	1
SPC	1.0E-01	8.0E-03	2.5E-09	5.0E-20	TBV8-02	1
SPC	1.0E-01	8.0E-03	4.6E-11	1.7E-23	TBV6-02	1
SPC	1.0E-01	8.0E-03	4.8E-07	1.8E-15	SERV-02	1
FWS	9.4E-01	7.2E-01	4.3E-09	1.3E-17	MFWRB	0
FWS	2.8E-02	6.3E-01	1.5E-05	1.4E-13	MFWRB	1
FWS	1.0E-01	8.0E-03	4.1E-07	1.4E-15	MFWRB	2
FWS	2.8E-02	6.3E-04	5.2E-08	1.7E-18	MFWRB	3
FWS	4.4E-03	1.6E-05	9.6E-07	1.4E-17	HLTRIP	1
FWS	5.6E-03	2.5E-05	1.2E-06	3.4E-17	EFW IN	1
FWS	1.4E-03	1.6E-06	1.2E-06	2.2E-18	EFW IN	2
FWS	3.4E-03	9.3E-06	3.6E-06	1.2E-16	EFW CONTL	1
FWS	1.0E-01	8.0E-03	9.4E-09	7.0E-19	EFW CONTL	2
ECC	1.3E-03	1.4E-06	2.9E-07	1.1E-19	HPI FAIL	1
ECC	4.0E-05	1.3E-09	0.0E+00	0.0E+00	HPI FAIL	1
PPC	1.0E-05	8.0E-11	5.7E-07	2.6E-23	PORV CLSD	1
PPC	5.0E-01	2.0E-01	4.2E-09	3.5E-18	BLCK VLVE	1
PPC	2.0E-02	3.2E-04	1.1E-06	3.5E-16	SRV RE	1
PPC	2.7E-02	5.83-04	4.6E-07	1.2E-16	PORV RE	1

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transients. Rather this procedure serves as a means of estimating an upper bound on the uncertainty attributable to branch points of small probability transients. The effect of initiator frequency uncertainty relative to these same transients has already been accounted for in the values for the initiator frequency contribution. Since no operator intervention was assumed for the residual transients, no uncertainty because of the human factor component is appropriate.

7.7 Human Factor Analysis

Because of the transient assignment procedure used, it was necessary to reduce the operator action trees down to the single operator action most significant to the transient. This reduction prevents a detailed estimation of the individual operator action comprising the various action trees. However, it is possible to obtain an estimate of the uncertainty in expected operator action. The human factor probabilities for these actions are $\leq$ 1.2E-2. Using this as a mean, and an uncertainty factor of 30, produced a contribution to the variance in ϕ(TWC) of 5.2E-13/RY^2.

7.8 Fracture Mechanics Analysis

The sensitivity coefficients and partial derivatives for the various fracture mechanics parameters were obtained by rerunning the fracture mechanics calculation with only a single parameter change. All parameters analyzed, with the exception of the flaw density, were assumed to be representable by normal distributions. The flaw density was assumed to be log-normal with a mean of 1.0 and a multiplicative factor of 300.

7.9 Thermal- Hydraulics Analysis

The partial derivative required by the uncertainty analysis for the three principal thermal-hydraulic variables (p, T, and h) was obtained in an analogous manner to that used for the fracture mechanics variables. Even though it appears that conservatisms were included in the thermal-hydraulic analyses, it was necessary to assume a normal distribution for all three thermal-hydraulic variables, since only limited information was available.

7.10 Summary and Discussion

Table 7.3 summarizes the results for all parameters examined as part of the uncertainty analysis. The estimated standard deviation in the frequency of vessel failure, $\sigma[\phi\,(TWC)]$ is 5E-5/RY. The expected value of ϕ (TWC) obtained in this study was 4.5E-6/RY, and together these two parameters characterize the dispersion and central tendency of the final distribution. They do not, however, identify the type of distribution. It is evident that the frequency of failure will be nonnegative. The standard deviation, however, is 10 times the estimated value. This would suggest a log-normal distribution, resulting in the confidence intervals listed in Table 7.4.

The amount of dispersion in the result attributable to the variance of each parameter has been calculated and is tabulated in Table 7.3. As indicated, the majority of uncertainties (60%) can be attributed to the uncertainty inherent in the temperature histories used to characterize the transients. This is due to the large partial derivative derived for this parameter (1.4E-6/RY per °C), coupled with the large assumed parameter uncertainty (σ_T = 28°C).

Table 7.3 Sensitivity/uncertainty results summary

Class	Parameter	Type	Mu	Variance	Partial	Contribution	Branch:	#
FREQ	RT	INIT	6.0E+00	2.0E+00	3.43-07	2.3E-13	TSVS	1
FREQ	EMFW	INIT	1.0E-01	8.0E-03	4.4E-06	1.5E-13	TBVS-01	1
FREQ	LSLB	INIT	1.0E-03	3.3E-07	6.6E-04	1.4E-13	TBVS-C1	2
FREQ	SSLB	INIT	1.0E-02	8.0E-05	7.8E-06	4.8E-15	SSRV-01	1
FREQ	LOMFW	INIT	5.0E-01	2.0E-01	8.5E-07	1.5E-13	TSVS-2	1
FREQ	SBL1	INIT	1.0E-01	8.0E-03	8.4E-06	5.7E-13	TBVS-02	1
FREQ	SBL2	INIT	1.0E-02	8.0E-05	1.1E-05	9.7E-15	TBVS-C2	1
FREQ	SGTR	INIT	1.0E-02	8.0E-05	1.0E-05	8.7E-15	SSRV-02	1
FREQ	SI	INIT	1.0E-02	8.0E-05	2.1E-05	3.6E-14	MFWRB	0
FREQ	BP	SPC	9.5E-04	2.4E-06	3.5E-07	2.9E-19	MFWRB	1
FREQ	BP	SPC	1.0E-03	2.6E-06	5.3E-06	7.4E-17	MFWRB	2
FREQ	BP	SPC	1.0E-04	2.6E-08	1.2E-06	3.6E-20	MFWRB	3
FREQ	BP	SPC	8.0E-02	1.7E-02	3.8E-06	2.4E-13	HLTRIP	1
FREQ	BP	SPC	1.0E-01	2.6E-02	2.2E-06	1.3E-13	EFW IN	1
FREQ	BP	SPC	1.0E-01	2.6E-02	2.5E-09	1.6E-19	EFW IN	2
FREQ	BP	SPC	1.0E-01	2.6E-02	4.6E-11	5.5E-23	EFW CONTL	1
FREQ	BP	SPC	1.0E-01	2.6E-02	4.8E-07	6.0E-15	EFW CONTL	2
FREQ	BP	FWS	9.4E-01	2.3E+00	4.3E-09	4.2E-17	HPI FAIL	1
FREQ	BP	FWS	2.8E-02	2.1E-03	1.5E-05	4.6E-13	CFT FAIL	1
FREQ	BP	FWS	1.0E-01	2.6E-02	4.1E-07	4.5E-15	PORV CLSD	1
FREQ	BP	FWS	2.8E-02	2.1E-03	5.2E-08	5.6E-18	BLCK VLVE	1
FREQ	BP	FWS	4.4E-03	5.1E-05	9.6E-07	4.6E-17	SRV RE	1
FREQ	BP	FWS	5.6E-03	8.2E-05	1.2E-06	1.1E-16	PORV RE	1

Table 7.3 Sensitivity/Uncertainty Results Summary (continued)

Class	Parameter	Type	Mu	Variance	Partial	Contribution	Branch: #
FREQ	BP	FWS	1.4E-03	5.1E-06	1.2E-06	7.3E-18	
FREQ	BP	FWS	3.4E-03	3.0E-05	3.6E-06	4.0E-16	
FREQ	BP	FWS	1.0E-01	2.6E-02	9.4E-09	2.3E-18	
FREQ	BP	ECC	1.3E-03	4.4E-06	2.9E-07	3.7E-19	
FREQ	BP	ECC	4.0E-05	4.2E-09	0.0E+00	0.0E+00	
FREQ	BP	PPC	1.0E-05	2.6E-10	5.7E-07	8.4E-23	
FREQ	BP	PPC	5.0E-01	6.5E-01	4.2E-09	1.2E-17	
FREQ	BP	PPC	2.0E-02	1.0E-03	1.1E-06	1.2E-15	
FREQ	BP	PPC	2.7E-02	1.9E-03	4.6E-07	4.0E-16	
FREQ	BP	RES		3.2E+00	1.2E-05	4.6E-10	
FREQ	BP	WF	1.0E-02	2.6E-04	6.0E-05	9.4E-13	
FM	RHOFLAW		1.0E+00	3.6E+01	4.5E-06	7.3E-10	
FM	FLUENCE		1.1E+00	1.1E-01	1.1E-05	1.3E-11	
FM	CU%		2.9E-01	6.3E-04	1.7E-04	1.8E-11	
FM	NI%		5.5E-01	0.0E+00	0.0E+00	0.0E+00	
FM	RTNDTO		-7.0E+00	8.1E+01	4.7E-07	1.8E-11	
FM	DELRTNDT		9.0E+01	8.8E+01	4.7E-07	1.9E-11	
FM	KIC		NA	2.3E-02	5.6E-05	7.1E-11	
FM	KIA		NA	1.0E-02	9.9E-07	9.8E-15	
TH	HTCOEFF		NA	6.3E-02	1.8E-06	2.0E-13	
TH	TEMP		NA	7.8E+02	1.4E-06	1.5E-09	
TH	PRESS		NA	1.2E-01	7.7E-07	6.9E-14	

TOTAL = 2.5×10^{-9}

Table 7.4 Confidence intervals for a lognormal assumption

Confidence Level	Factor	Lower Bound	Upper Bound
68%	9.0	5.0E-07	4.1E-05
95%	82	5.5E-08	3.7E-04
99.7%	737	6.1E-09	3.3E-03

The second largest contribution (29%) to the overall uncertainty is that attributable to the flaw density in the weld. Once again, the magnitude of the contribution is due to a large partial derivative coupled with a large parameter uncertainty.

Ranking third (18%) as a source of uncertainty is the contribution due to the residual transient uncertainties. This contribution is attributable to two factors. First, the variance estimated for the branch points is large as a result of the conservative procedure utilized to obtain it. Second, the partial derivative calculated for this parameter incorporates the conditional probability of failure $[P(F/E_i)]$ calculated for the default transient (the highest conditional failure probability obtained for any of the transients).

Finally, it is useful to sum the contributions of the various parameters into three classes representing the three major influences on frequency of vessel failure (event frequency, fracture mechanics, or thermal hydraulics). This breakdown is given in Table 7.5.

Table 7.5 Uncertainty contribution by major parameter classification

Classification	Contribution	Correlations
Event frequency	1.4E-10	0.056
Fracture mechanics	8.7E-10	0.34
Thermal hydraulics	1.5E-09	0.60
Total	2.5E-9	1.00

8.0 CONCLUSIONS AND RECOMMENDATIONS

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8.1 Introduction

This chapter consists of specific and general conclusions which were drawn from the results of the report and also from various aspects of the analysis. In addition, Section 8.3 discusses areas of the report that require further study and development if a similar Pressurized Thermal Shock (PTS) analysis were to be performed on other nuclear plants.

8.2 Conclusions from the Oconee-1 Study

8.2.1 Oconee-1 System Features and Proposed System Changes

Major systems related to the Oconee-1 PTS analysis were described in Chapter 2. From the analysis it became apparent that two of the Oconee-1 system features play major roles in reducing the through-the-wall crack (TWC) risk:

(1) The vent valves in the B&W reactor vessel minimize thermal stratification in the downcomer region when primary loop flow stagnates.

(2) The steam generator at Oconee-1 has a high level trip feature (installed by Duke Power) which prevents overfill on MFW. Without such a feature, the overall risk of a TWC increases by 35%. This increase was based on the assumption that main feedwater flow could be sustained after

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the steam generator filled with water. It appears that there are some aspects of the design of the system that might invalidate this assumption and thus reduce the increase in TWC risk. One of these aspects is that the main feedwater pumps are driven by steam turbines which extract their steam from the lowest point at the end of the steam line. Water entrained in the steam would result in loss of turbine power to the feedwater pumps and subsequently result in a termination of the feedwater event. Secondly, the water dynamics in the steam line might lead to failure in the steam line which would probably result in a blowdown instead of an overfeed. These effects were not analyzed in this study and thus cannot be quantified.

Two system changes have been examined which could lead to additional reduction of the risk of TWC:

(1) The first change involves the heating of the high-pressure injection (HPI) water by 50°F. Since complete mixing occurs during most of the transients in Oconee-1, the actual downcomer water temperature is estimated to increase less than 10°F (well within the estimated temperature uncertainty of this study), thereby providing negligible benefit.

(2) A second system change is a modification of the control or operational procedures so as to limit the repressurization of the primary system during an overcooling transient. If such a system were 100% effective, the risk of a TWC would decrease at least a factor of 14 to 3.2×10^{-7}/RY. This could most easily be accomplished by appropriate throttling of the HPI. (This throttling was not taken into account in the Oconee-1 study.) It should be noted that Duke Power is currently implementing the

anticipated transient operating guideline (ATOG) program which includes operator action to limit repressurization as part of the response to overcooling events.

8.2.2 Accident Sequence Analysis

The accident sequence analysis is discussed in Chapters 3 and 6 of this report and several significant conclusions can be drawn:

(1) The results show that the risk profile is dominated by the "residual risk."* The three reasons for this large residual risk are as follows: (a) No operator action trees were assigned to any sequence in the "residual risk" category; thus the human error probability was 1.0. (b) A single severe thermal-hydraulic transient was assigned to each event sequence in the residual category, which resulted in a constant conditional probability of TWC of 5.4×10^{-3}. (c) There was a large number of event sequences (several million) which were lumped into the "residual risk" category. Therefore, it is our opinion that more selective techniques should be considered for assigning transients to the residual risk category in future studies.

(2) In assessing the overall risk of TWC in Oconee-1, no individual sequence, when combined with the conditional probability of a TWC, had a frequency larger than 6×10^{-7}/RY.

(3) The steam line break initiators were found to be the most important contributors to risk (second only to the residual in total risk contributions).

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*All event sequences whose frequencies were less than 10^{-6}/RY were lumped into a category called "residual risk."

(4) Of the sequences analyzed, those with high frequency of occurrence have low TWC conditional probability. Conversely, events which have high TWC conditional probability have low frequency of occurrence.

8.2.3 Fracture Mechanics Analysis

From the fracture mechanics analysis in Chapter 5, several conclusions can be drawn:

(1) The axial welds (Weld Numbers SA1073, SA1430, and SA1493) dominate the contribution to the TWC conditional probability. Assuming the same flaw density throughout the beltline region of the vessel, the circumferential welds would add approximately 5% to P(F/E) and the base material approximately 30%.

(2) Aside from the flaw density, P(F/E) is by far most sensitive to uncertainties in the temperature of the coolant in the downcomer and least sensitive to the fluid-film heat transfer coefficient at the vessel inner surface and the primary-system pressure.

(3) A reduction in the fluence rate could have a significant impact on the risk of a TWC, but a cost/benefit analysis is needed.

(4) Improved in-service inspection followed by repair of all flaws that are found could reduce the PTS risk significantly, but complete repair and precise detection of flaws are difficult with today's technology.

(5) Annealing the pressure vessel would significantly reduce the PTS risk, but the cost and uncertainty regarding technical feasibility probably render this a last resort fix to the PTS problem.

(6) The inclusion of WPS in the FM analysis reduces P(F/E) by several orders of magnitude for many, but not all, of the idealized transients. Because of concerns over the applicability of WPS under actual transient conditions, it was not included in the estimate of (TWC), but this does not mean that conditions for which WPS would be effective cannot be expected to occur.

(7) Limiting the primary-system pressure to approximately 7 MPa (1000 psi) during repressurization for those transients that would otherwise experience higher pressures reduced P(F/E) by as much as two orders of magnitude for individual transients.

(8) For many of the transients, the predicted failures occurred near the end of the assumed 2-h duration (see Appendix E). This is approaching a cold overpressurization situation, and thus extending the duration of the transient could substantially increase P(F/E). Conversely, if the duration were decreased, P(F/E) would be decreased.

8.2.4 Uncertainty and Sensitivity Analysis

The uncertainty and sensitivity analysis reported in Chapter 7 indicates the following:

(1) The uncertainty in the temperature of the downcomer fluid (±28°C) dominates the uncertainty in frequency of a TWC.

(2) The uncertainty in the flaw density is the largest uncertainty of any factor considered and results in the second largest uncertainty contribution to the TWC frequency.

(3) The failure of the operator to isolate the steam generator during a steam line break represents, with the exception of the failure to throttle HPI, the largest human error contribution to the frequency of a TWC.

8.2.5 General Conclusions

In addition to the specific conclusions drawn from the specific chapters of the study, the following general statements can be made:

(1) Only minimal system interactions, no external events (fires, floods, earthquakes, or sabotage) and no inadvertent operator actions were considered in the Oconee-1 study. The impact of these effects on the frequency of a TWC cannot be estimated either qualitatively or quantitatively without further analysis.

(2) The best estimate of the frequency of a TWC (4.5×10^{-6}/RY) is small* when compared to the proposed safety goal. This number is about a

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*This assumes that the conditional probability of core melt, given a TWC, is 1.

factor of 10 less than the dominant core melt probabilities calculated in plant-specific PRAs, which implies that PTS is a small contributor to overall risk of core melt.

(3) The best estimate of the frequency of a TWC based on information utilized in this report is nearly six times larger than the upper limit (8.5×10^{-7}/RY) reported in the B&W Owners Group Study.*

8.3 Areas Requiring Further Study and Development

8.3.1 Human Reliability

The method used for quantification of human error in the Oconee-1 study clearly needs improvement. A more sophisticated technique was adopted for later studies performed for the Calvert Cliffs-1 and H. B. Robinson-2 plants. As in all PRAs, this subject deserves considerable attention. The operator obviously plays a significant role in the progress of a PTS sequence, but because of the simplistic treatment in this study, the calculated sensitivity of the TWC to operator action is probably not as large as would occur if a more sophisticated approach to human action were to be implemented.

8.3.2 System Interactions

The approach used to evaluate system interactions for this study was also simplistic and no system interactions were identified in the Oconee-1 analysis. More in-depth analyses for the Calvert Cliffs-1 and H. B. Robinson-2 studies did indicate the existence of system interactions, which could prove important to PTS sequences in these plants.

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*B&W Owners Group, "Probability Evaluation of Pressurized Thermal Shock Phase I," pp. 2-3, BAW-1791 (June 1983).

8.3.3 External Events

The frequency of external events and their impact on PTS was excluded from the Oconee-1 study and from the Calvert Cliffs-1 and H. B. Robinson-2 studies as well. Since many of the significant sequences in Oconee-1 involve multiple failures, it is possible that an external event could increase the frequency of the dominant sequences by eliminating the randomness of the failures. However, most external event analyses are found to affect the cooling systems and control systems of reactors and one might expect an undercooling event due to a external hazard to be more prevalent than an overcooling event.

8.3.4 Flooding

Late in the study, the subject of flooding of the reactor cavity surrounding the vessel was discussed; however, neither time nor money was available to analyze this scenario.

8.3.5 Thermal-Hydraulic Modeling

It is clear that the thermal-hydraulic (T-H) modeling requires considerable effort in building and validating the primary and secondary models, especially in conjunction with an integrated control system such as appears in the Oconee-1 reactor. Even when the plant systems are purported to be correctly modeled, the RELAP and TRAC codes do not give identical answers, and in one case, the predicted downcomer temperatures differed by more than 140°F. The effect of modeling should be carefully addressed whenever a PTS study is done.

8.3.6 Decay Heat Assumptions

Because of the vent valves in Oconee-1, loop stagnation does not appear to be a problem for loss-of-coolant accidents (LOCA). However, should a LOCA occur at hot standby following an extended shutdown, stagnation is likely to occur and might lead to low downcomer temperatures. Also, without a source of decay heat, a steam line break at hot standby could cause significantly lower downcomer temperatures. On the other hand, the frequency of these events is lower than similar events at full power, so the effect on the total risk should be examined in future studies.

8.3.7 Duration of Calculated Transients

Throughout the study the T-H analysis was arbitrarily terminated after two hours, the underlying assumption being that some operator action would be taken by that time to mitigate the event. However, for non-isolatable events (LOCAs and steam line breaks), this may not be a realistic assumption. Furthermore, it appears that for several transients a large percentage of TWCs occur at the two hour limit. The effect of the two hour limitation merits further analysis in light of the relative importance of the choice of this time period.

8.4 Summary

Although this study does not completely eliminate pressurized thermal shock as an unresolved safety issue for the Oconee-1 plant, it does provide a thorough study of the effect of various potential overcooling transients on the reactor vessel. It is realized that much of what is stated in this report was known to some extent and in piecemeal fashion by various technical specialists prior to the study. However, by integrating the disciplines of probabilistic risk analysis, thermal-hydraulic analysis, and fracture mechanics, and by adopting a consistent technique of assessing uncertainties and sensitivities across these disciplines, a clearer understanding of the total aspect of the pressurized thermal shock problem has resulted. In particular, the uncertainty analysis, although far from perfect, represents an attempt to adopt a consistent and mathematically sound analysis of the problem. Such an analysis should be a requisite for any future pressurized thermal shock study performed by NRC or a utility. Even with the deficiencies outlined in Section 8.3, the results of this study should be of use to both NRC and Duke Power Company in resolving the plant-specific aspects of pressurized thermal shock. The approach taken in the Oconee-1 study (with modifications) should provide a basis for future plant-specific probabilistic risk assessments of pressurized thermal shock.

APPENDIX A

EVENT TREES

The computerized event trees are extensive and too large to be included in this report. They are available from George Flanagan, Engineering Physics and Mathematics Division, Building 6025, Oak Ridge National Laboratory, P.O. Box X, Oak Ridge, TN 37831.

Three sets of data are presented in this appendix; Section A.1 is a log of sequences by identification number and frequency for system state tree branch points, Section A.2 is a log of sequences summed by system state category, and Section A.3 relates the combination of end state frequencies of Section A.2 operator action probabilities and fracture-mechanics probabilities of Table 6.1.

A.1 Sequences by Identification Number and Frequency, System State Tree Branch Points

			--SYSTEM STATE BRANCHES----					----------CATEGORIES----------			
REC	INIT	SEQ#	SPC	FWS	ECC	PPC	EFREQ	SPC	FWS	ECC	PPC
1	RT	6556	1	24	1	1	4.8E+00	0	0	0	0
2	RT	6841	1	25	1	1	1.4E-01	0	1	0	0
3	RT	7148	1	26	2	8	2.0E-04	0	3	0	0
4	RT	7149	1	26	2	9	5.5E-06	0	3	0	1
5	RT	7150	1	26	2	10	1.3E-05	0	3	0	2
6	RT	7152	1	26	2	12	2.3E-05	0	3	0	0
7	RT	7153	1	26	2	13	1.4E-06	0	3	0	2
8	RT	7433	1	27	2	8	1.2E-05	0	3	0	0
9	RT	7437	1	27	2	12	1.3E-06	0	3	0	0
10	RT	7703	1	28	1	8	6.6E-04	0	4	0	0
11	RT	7704	1	28	1	9	1.8E-05	0	4	0	1
12	RT	7705	1	28	1	10	4.2E-05	0	4	0	2
13	RT	7706	1	28	1	11	1.2E-06	0	4	0	2
14	RT	7707	1	28	1	12	7.5E-05	0	4	0	0
15	RT	7708	1	28	1	13	4.8E-06	0	4	0	2
16	RT	7988	1	29	1	8	1.7E-04	0	4	0	0
17	RT	7989	1	29	1	9	4.6E-06	0	4	0	1
18	RT	7990	1	29	1	10	1.1E-05	0	4	0	2
19	RT	7992	1	29	1	12	1.9E-05	0	4	0	0
20	RT	7993	1	29	1	13	1.2E-06	0	4	0	2
21	RT	8288	1	30	2	8	5.2E-04	0	2	0	0
22	RT	8289	1	30	2	9	1.4E-05	0	2	0	1
23	RT	8290	1	30	2	10	3.3E-05	0	2	0	2
24	RT	8292	1	30	2	12	5.9E-05	0	2	0	0
25	RT	8293	1	30	2	13	3.6E-06	0	2	0	2
26	RT	8551	1	31	1	1	1.4E-02	0	1	0	0
27	RT	8858	1	32	2	8	2.0E-05	0	3	0	0
28	RT	8860	1	32	2	10	1.3E-06	0	3	0	2
29	RT	8862	1	32	2	12	2.3E-06	0	3	0	0
30	RT	9143	1	33	2	8	1.2E-06	0	3	0	0
31	RT	9413	1	34	1	8	6.6E-05	0	4	0	0
32	RT	9414	1	34	1	9	1.8E-06	0	4	0	1
33	RT	9415	1	34	1	10	4.2E-06	0	4	0	2
34	RT	9417	1	34	1	12	7.5E-06	0	4	0	0
35	RT	9698	1	35	1	8	1.7E-05	0	4	0	0
36	RT	9700	1	35	1	10	1.1E-05	0	4	0	2
37	RT	9702	1	35	1	12	1.9E-06	0	4	0	0
38	RT	9998	1	36	2	8	1.2E-06	0	2	0	0
39	RT	10261	1	37	1	1	1.4E-01	0	0	0	0
40	RT	10568	1	38	2	8	2.0E-04	0	3	0	0
41	RT	10569	1	38	2	9	5.5E-06	0	3	0	1
42	RT	10570	1	38	2	10	1.3E-05	0	3	0	2
43	RT	10572	1	38	2	12	2.3E-05	0	3	0	0
44	RT	10573	1	38	2	13	1.4E-06	0	3	0	2
45	RT	10853	1	39	2	8	1.2E-05	0	3	0	0
46	RT	10857	1	39	2	12	1.3E-06	0	3	0	0
47	RT	11123	1	40	1	8	6.6E-04	0	4	0	0
48	RT	11124	1	40	1	9	1.8E-05	0	4	0	1
49	RT	11125	1	40	1	10	4.2E-05	0	4	0	2

			--SYSTEM STATE BRANCHES----					----------CATEGORIES----------			
REC	INIT	SEQ#	SPC	FWS	ECC	PPC	EFREQ	SPC	FWS	ECC	PPC
50	RT	11126	1	40	1	11	1.2E-06	0	4	0	2
51	RT	11127	1	40	1	12	7.5E-05	0	4	0	0
52	RT	11128	1	40	1	13	4.8E-06	0	4	0	2
53	RT	11408	1	41	1	8	1.7E-04	0	4	0	0
54	RT	11409	1	41	1	9	4.6E-06	0	4	0	1
55	RT	11410	1	41	1	10	1.1E-05	0	4	0	2
56	RT	11412	1	41	1	12	1.9E-05	0	4	0	0
57	RT	11413	1	41	1	13	1.2E-06	0	4	0	2
58	RT	74978	2	95	2	8	3.0E-01	1	0	0	0
59	RT	74979	2	95	2	9	8.3E-03	1	0	0	1
60	RT	74980	2	95	2	10	1.9E-02	1	0	0	2
61	RT	74981	2	95	2	11	5.3E-04	1	0	0	2
62	RT	74982	2	95	2	12	3.4E-02	1	0	0	0
63	RT	74983	2	95	2	13	2.2E-03	1	0	0	2
64	RT	74984	2	95	2	14	1.0E-04	1	0	0	0
65	RT	74985	2	95	2	15	6.5E-06	1	0	0	2
66	RT	74993	2	95	3	8	3.9E-04	1	0	1	0
67	RT	74994	2	95	3	9	1.1E-05	1	0	1	1
68	RT	74995	2	95	3	10	2.5E-05	1	0	1	2
69	RT	74997	2	95	3	12	4.4E-05	1	0	1	0
70	RT	74998	2	95	3	13	2.8E-06	1	0	1	2
71	RT	75263	2	96	2	8	5.1E-04	1	3	0	0
72	RT	75264	2	96	2	9	1.4E-05	1	3	0	1
73	RT	75265	2	96	2	10	3.3E-05	1	3	0	2
74	RT	75267	2	96	2	12	5.8E-05	1	3	0	0
75	RT	75268	2	96	2	13	3.7E-06	1	3	0	2
76	RT	75548	2	97	2	8	3.0E-05	1	3	0	0
77	RT	75550	2	97	2	10	1.9E-06	1	3	0	2
78	RT	75552	2	97	2	12	3.4E-06	1	3	0	0
79	RT	75833	2	98	2	8	1.7E-03	1	4	0	0
80	RT	75834	2	98	2	9	4.7E-05	1	4	0	1
81	RT	75835	2	98	2	10	1.1E-04	1	4	0	2
82	RT	75836	2	98	2	11	3.0E-06	1	4	0	2
83	RT	75837	2	98	2	12	1.9E-04	1	4	0	0
84	RT	75838	2	98	2	13	1.2E-05	1	4	0	2
85	RT	75848	2	98	3	8	2.2E-06	1	4	1	0
86	RT	76118	2	99	2	8	4.3E-04	1	4	0	0
87	RT	76119	2	99	2	9	1.2E-05	1	4	0	1
88	RT	76120	2	99	2	10	2.7E-05	1	4	0	2
89	RT	76122	2	99	2	12	4.8E-05	1	4	0	0
90	RT	76123	2	99	2	13	3.1E-06	1	4	0	2
91	RT	76403	2	100	2	8	8.9E-03	1	1	0	0
92	RT	76404	2	100	2	9	2.5E-04	1	1	0	1
93	RT	76405	2	100	2	10	5.7E-04	1	1	0	2
94	RT	76406	2	100	2	11	1.6E-05	1	1	0	2
95	RT	76407	2	100	2	12	1.0E-03	1	1	0	0
96	RT	76408	2	100	2	13	6.5E-05	1	1	0	2
97	RT	76409	2	100	2	14	3.0E-06	1	1	0	0
98	RT	76418	2	100	3	8	1.2E-05	1	1	1	0

			--SYSTEM STATE BRANCHES----					----------CATEGORIES----------			
REC	INIT	SEQ#	SPC	FWS	ECC	PPC	EFREQ	SPC	FWS	ECC	PPC
393	RT	1760755	37	95	2	10	1.7E-05	1	0	0	2
394	RT	1760757	37	95	2	12	3.1E-05	1	0	0	0
395	RT	1760758	37	95	2	13	2.0E-06	1	0	0	2
396	RT	1761608	37	98	2	8	1.5E-06	1	4	0	0
397	RT	1762178	37	100	2	8	8.1E-06	1	1	0	0
398	RT	1767878	37	120	2	8	8.1E-06	1	0	0	0
399	RT	1808918	38	95	2	8	6.0E-05	3	0	0	0
400	RT	1808919	38	95	2	9	1.6E-06	3	0	0	1
401	RT	1808920	38	95	2	10	3.8E-06	3	0	0	2
402	RT	1808922	38	95	2	12	6.8E-06	3	0	0	0
403	RT	1810343	38	100	2	8	1.8E-06	3	1	0	0
404	RT	1816043	38	120	2	8	1.8E-06	3	0	0	0
405	RT	1857083	39	95	2	8	3.0E-05	4	0	0	0
406	RT	1857085	39	95	2	10	1.9E-06	4	0	0	2
407	RT	1857087	39	95	2	12	3.4E-06	4	0	0	0
408	RT	1905248	40	95	2	8	6.7E-06	3	0	0	0
409	RT	2318476	49	24	1	1	4.3E-04	0	0	0	0
410	RT	2318761	49	25	1	1	1.3E-05	0	1	0	0
411	RT	2320471	49	31	1	1	1.3E-06	0	1	0	0
412	RT	2322181	49	37	1	1	1.3E-05	0	0	0	0
413	RT	2386899	50	95	2	8	2.7E-05	1	0	0	0
414	RT	2386900	50	95	2	10	1.7E-06	1	0	0	2
415	RT	2386902	50	95	2	12	3.1E-06	1	0	0	0
416	RT	2511136	53	24	1	1	4.8E-05	0	0	0	0
417	RT	2511421	53	25	1	1	1.4E-06	0	1	0	0
418	RT	2514841	53	37	1	1	1.4E-06	0	0	0	0
419	RT	2579538	54	95	2	8	3.0E-06	1	0	0	0
420	RT	2916713	61	95	2	8	2.7E-05	1	0	0	0
421	RT	2916715	61	95	2	10	1.7E-06	1	0	0	2
422	RT	2916717	61	95	2	12	3.1E-06	1	0	0	0
423	RT	2964878	62	95	2	8	6.0E-06	3	0	0	0
424	RT	3109373	65	95	2	8	3.0E-06	1	0	0	0
425	RT	6364336	133	24	1	1	4.5E-03	0	0	0	0
426	RT	6364621	133	25	1	1	1.3E-04	0	1	0	0
427	RT	6366331	133	31	1	1	1.3E-05	0	1	0	0
428	RT	6368041	133	37	1	1	1.3E-04	0	0	0	0
429	RT	6432758	134	95	2	8	2.8E-04	1	0	0	0
430	RT	6432759	134	95	2	9	7.7E-06	1	0	0	1
431	RT	6432760	134	95	2	10	1.8E-05	1	0	0	2
432	RT	6432762	134	95	2	12	3.2E-05	1	0	0	0
433	RT	6432763	134	95	2	13	2.0E-06	1	0	0	2
434	RT	6433613	134	98	2	8	1.6E-06	1	4	0	0
435	RT	6434183	134	100	2	8	8.3E-06	1	1	0	0
436	RT	6439883	134	120	2	8	8.4E-06	1	0	0	0
437	RT	6480923	135	95	2	8	3.3E-06	2	0	0	0
438	RT	6914408	144	95	2	8	4.4E-04	6	0	0	0
439	RT	6914409	144	95	2	9	1.2E-05	6	0	0	1
440	RT	6914410	144	95	2	10	2.8E-05	6	0	0	2
441	RT	6914412	144	95	2	12	5.0E-05	6	0	0	0

REC	INIT	SEQ#	--SYSTEM STATE BRANCHES---- SPC	FWS	ECC	PPC	EFREQ	----------CATEGORIES---------- SPC	FWS	ECC	PPC
99	RT	76422	2	100	3	12	1.3E-06	1	1	1	0
100	RT	76688	2	101	2	8	1.5E-05	1	3	0	0
101	RT	76692	2	101	2	12	1.7E-06	1	3	0	0
102	RT	77258	2	103	2	8	5.0E-05	1	4	0	0
103	RT	77259	2	103	2	9	1.4E-06	1	4	0	1
104	RT	77260	2	103	2	10	3.2E-06	1	4	0	2
105	RT	77262	2	103	2	12	5.7E-06	1	4	0	0
106	RT	77543	2	104	2	8	1.3E-05	1	4	0	0
107	RT	77547	2	104	2	12	1.4E-06	1	4	0	0
108	RT	77828	2	105	2	8	3.9E-05	1	2	0	0
109	RT	77829	2	105	2	9	1.1E-06	1	2	0	1
110	RT	77830	2	105	2	10	2.5E-06	1	2	0	2
111	RT	77832	2	105	2	12	4.5E-06	1	2	0	0
112	RT	79253	2	110	2	8	9.0E-04	1	1	0	0
113	RT	79254	2	110	2	9	2.5E-05	1	1	0	1
114	RT	79255	2	110	2	10	5.7E-05	1	1	0	2
115	RT	79256	2	110	2	11	1.6E-06	1	1	0	2
116	RT	79257	2	110	2	12	1.0E-04	1	1	0	0
117	RT	79258	2	110	2	13	6.5E-06	1	1	0	2
118	RT	79268	2	110	3	8	1.2E-06	1	1	1	0
119	RT	79538	2	111	2	8	1.5E-06	1	3	0	0
120	RT	80108	2	113	2	8	5.1E-06	1	4	0	0
121	RT	80393	2	114	2	8	1.3E-06	1	4	0	0
122	RT	82103	2	120	2	8	9.0E-03	1	0	0	0
123	RT	82104	2	120	2	9	2.5E-04	1	0	0	1
124	RT	82105	2	120	2	10	5.7E-04	1	0	0	2
125	RT	82106	2	120	2	11	1.6E-05	1	0	0	2
126	RT	82107	2	120	2	12	1.0E-03	1	0	0	0
127	RT	82108	2	120	2	13	6.5E-05	1	0	0	2
128	RT	82109	2	120	2	14	3.1E-06	1	0	0	0
129	RT	82118	2	120	3	8	1.2E-05	1	0	1	0
130	RT	82122	2	120	3	12	1.3E-06	1	0	1	0
131	RT	82388	2	121	2	8	1.5E-05	1	3	0	0
132	RT	82392	2	121	2	12	1.7E-06	1	3	0	0
133	RT	82958	2	123	2	8	5.1E-05	1	4	0	0
134	RT	82959	2	123	2	9	1.4E-06	1	4	0	1
135	RT	82960	2	123	2	10	3.2E-06	1	4	0	2
136	RT	82962	2	123	2	12	5.7E-06	1	4	0	0
137	RT	83243	2	124	2	8	1.3E-05	1	4	0	0
138	RT	83247	2	124	2	12	1.4E-06	1	4	0	0
139	RT	123143	3	95	2	8	3.5E-03	2	0	0	0
140	RT	123144	3	95	2	9	9.8E-05	2	0	0	1
141	RT	123145	3	95	2	10	2.3E-04	2	0	0	2
142	RT	123146	3	95	2	11	6.2E-06	2	0	0	2
143	RT	123147	3	95	2	12	4.0E-04	2	0	0	0
144	RT	123148	3	95	2	13	2.6E-05	2	0	0	2
145	RT	123149	3	95	2	14	1.2E-06	2	0	0	0
146	RT	123158	3	95	3	8	4.6E-06	2	0	1	0
147	RT	123428	3	96	2	8	6.0E-06	2	3	0	0

			--SYSTEM STATE BRANCHES----					----------CATEGORIES----------			
REC	INIT	SEQ#	SPC	FWS	ECC	PPC	EFREQ	SPC	FWS	ECC	PPC
148	RT	123998	3	98	2	8	2.0E-05	2	4	0	0
149	RT	124000	3	98	2	10	1.3E-06	2	4	0	2
150	RT	124002	3	98	2	12	2.3E-06	2	4	0	0
151	RT	124283	3	99	2	8	5.0E-06	2	4	0	0
152	RT	124568	3	100	2	8	1.1E-04	2	1	0	0
153	RT	124569	3	100	2	9	2.9E-06	2	1	0	1
154	RT	124570	3	100	2	10	6.7E-06	2	1	0	2
155	RT	124572	3	100	2	12	1.2E-05	2	1	0	0
156	RT	127418	3	110	2	8	1.1E-05	2	1	0	0
157	RT	127422	3	110	2	12	1.2E-06	2	1	0	0
158	RT	130268	3	120	2	8	1.1E-04	2	0	0	0
159	RT	130269	3	120	2	9	2.9E-06	2	0	0	1
160	RT	130270	3	120	2	10	6.7E-06	2	0	0	2
161	RT	130272	3	120	2	12	1.2E-05	2	0	0	0
162	RT	171308	4	95	2	8	2.7E-04	1	0	0	0
163	RT	171309	4	95	2	9	7.5E-06	1	0	0	1
164	RT	171310	4	95	2	10	1.7E-05	1	0	0	2
165	RT	171312	4	95	2	12	3.1E-05	1	0	0	0
166	RT	171313	4	95	2	13	2.0E-06	1	0	0	2
167	RT	172163	4	98	2	8	1.5E-06	1	4	0	0
168	RT	172733	4	100	2	8	8.0E-06	1	1	0	0
169	RT	178433	4	120	2	8	6.1E-06	1	0	0	0
170	RT	199216	5	24	1	1	4.3E-04	0	0	0	0
171	RT	199501	5	25	1	1	1.3E-05	0	1	0	0
172	RT	201211	5	31	1	1	1.3E-06	0	1	0	0
173	RT	202921	5	37	1	1	1.3E-05	0	0	0	0
174	RT	267638	6	95	2	8	2.7E-05	1	0	0	0
175	RT	267640	6	95	2	10	1.7E-06	1	0	0	2
176	RT	267642	6	95	2	12	3.1E-06	1	0	0	0
177	RT	536371	12	24	1	1	4.4E-03	0	0	0	0
178	RT	536656	12	25	1	1	1.3E-04	0	1	0	0
179	RT	538366	12	31	1	1	1.3E-05	0	1	0	0
180	RT	540076	12	37	1	1	1.3E-04	0	0	0	0
181	RT	604793	13	95	2	8	3.0E-01	1	0	0	0
182	RT	604794	13	95	2	9	8.3E-03	1	0	0	1
183	RT	604795	13	95	2	10	1.9E-02	1	0	0	2
184	RT	604796	13	95	2	11	5.3E-04	1	0	0	2
185	RT	604797	13	95	2	12	3.4E-02	1	0	0	0
186	RT	604798	13	95	2	13	2.2E-03	1	0	0	2
187	RT	604799	13	95	2	14	1.0E-04	1	0	0	0
188	RT	604800	13	95	2	15	6.6E-06	1	0	0	2
189	RT	604808	13	95	3	8	3.9E-04	1	0	1	0
190	RT	604909	13	95	3	9	1.1E-05	1	0	1	1
191	RT	604810	13	95	3	10	2.5E-05	1	0	1	2
192	RT	604812	13	95	3	12	4.5E-05	1	0	1	0
193	RT	604813	13	95	3	13	2.8E-06	1	0	1	2
194	RT	605078	13	96	2	8	5.2E-04	1	3	0	0
195	RT	605079	13	96	2	9	1.4E-05	1	3	0	1
196	RT	605080	13	96	2	10	3.3E-05	1	3	0	2

			--SYSTEM STATE BRANCHES----					----------CATEGORIES----------			
REC	INIT	SEQ#	SPC	FWS	ECC	PPC	EFREQ	SPC	FWS	ECC	PPC
197	RT	605082	13	96	2	12	5.8E-05	1	3	0	0
198	RT	605083	13	96	2	13	3.7E-06	1	3	0	2
199	RT	605363	13	97	2	8	3.0E-05	1	3	0	0
200	RT	605365	13	97	2	10	1.9E-06	1	3	0	2
201	RT	605367	13	97	2	12	3.4E-06	1	3	0	0
202	RT	605648	13	98	2	8	1.7E-03	1	4	0	0
203	RT	605649	13	98	2	9	4.7E-05	1	4	0	1
204	RT	605650	13	98	2	10	1.1E-04	1	4	0	2
205	RT	605651	13	98	2	11	3.0E-06	1	4	0	2
206	RT	605652	13	98	2	12	1.9E-04	1	4	0	0
207	RT	605653	13	98	2	13	1.2E-05	1	4	0	2
208	RT	605663	13	98	3	8	2.2E-06	1	4	1	0
209	RT	605933	13	99	2	8	4.3E-04	1	4	0	0
210	RT	605934	13	99	2	9	1.2E-05	1	4	0	1
211	RT	605935	13	99	2	10	2.7E-05	1	4	0	2
212	RT	605937	13	99	2	12	4.8E-05	1	4	0	0
213	RT	605938	13	99	2	13	3.1E-06	1	4	0	2
214	RT	606218	13	100	2	8	9.0E-03	1	1	0	0
215	RT	606219	13	100	2	9	2.5E-04	1	1	0	1
216	RT	606220	13	100	2	10	5.7E-04	1	1	0	2
217	RT	606221	13	100	2	11	1.6E-05	1	1	0	2
218	RT	606222	13	100	2	12	1.0E-03	1	1	0	0
219	RT	606223	13	100	2	13	6.5E-05	1	1	0	2
220	RT	606224	13	100	2	14	3.1E-06	1	1	0	0
221	RT	606233	13	100	3	8	1.2E-05	1	1	1	0
222	RT	606237	13	100	3	12	1.3E-06	1	1	1	0
223	RT	606503	13	101	2	8	1.5E-05	1	3	0	0
224	RT	606507	13	101	2	12	1.7E-06	1	3	0	0
225	RT	607073	13	103	2	8	5.1E-05	1	4	0	0
226	RT	607074	13	103	2	9	1.4E-06	1	4	0	1
227	RT	607075	13	103	2	10	3.2E-06	1	4	0	2
228	RT	607077	13	103	2	12	5.7E-06	1	4	0	0
229	RT	607358	13	104	2	8	1.3E-05	1	4	0	0
230	RT	607362	13	104	2	12	1.4E-06	1	4	0	0
231	RT	607643	13	105	2	8	4.0E-05	1	2	0	0
232	RT	607644	13	105	2	9	1.1E-06	1	2	0	1
233	RT	607645	13	105	2	10	2.5E-06	1	2	0	2
234	RT	607647	13	105	2	12	4.5E-06	1	2	0	0
235	RT	609068	13	110	2	8	9.0E-04	1	1	0	0
236	RT	609069	13	110	2	9	2.5E-05	1	1	0	1
237	RT	609070	13	110	2	10	5.7E-05	1	1	0	2
238	RT	609071	13	110	2	11	1.6E-06	1	1	0	2
239	RT	609072	13	110	2	12	1.0E-04	1	1	0	0
240	RT	609073	13	110	2	13	6.5E-06	1	1	0	2
241	RT	609083	13	110	3	8	1.2E-06	1	1	1	0
242	RT	609353	13	111	2	8	1.5E-06	1	3	0	0
243	RT	609923	13	113	2	8	5.1E-06	1	4	0	0
244	RT	610208	13	114	2	8	1.3E-06	1	4	0	0
245	RT	611918	13	120	2	8	9.0E-03	1	0	0	0

			--SYSTEM STATE BRANCHES----					----------CATEGORIES----------			
REC	INIT	SEQ#	SPC	FWS	ECC	PPC	EFREQ	SPC	FWS	ECC	PPC
246	RT	611919	13	120	2	9	2.5E-04	1	0	0	1
247	RT	611920	13	120	2	10	5.7E-04	1	0	0	2
248	RT	611921	13	120	2	11	1.6E-05	1	0	0	2
249	RT	611922	13	120	2	12	1.0E-03	1	0	0	0
250	RT	611923	13	120	2	13	6.5E-05	1	0	0	2
251	RT	611924	13	120	2	14	3.1E-06	1	0	0	0
252	RT	611933	13	120	3	8	1.2E-05	1	0	1	0
253	RT	611937	13	120	3	12	1.3E-06	1	0	1	0
254	RT	612203	13	121	2	8	1.5E-05	1	3	0	0
255	RT	612207	13	121	2	12	1.7E-06	1	3	0	0
256	RT	612773	13	123	2	8	5.1E-05	1	4	0	0
257	RT	612774	13	123	2	9	1.4E-06	1	4	0	1
258	RT	612775	13	123	2	10	3.2E-06	1	4	0	2
259	RT	612777	13	123	2	12	5.8E-06	1	4	0	0
260	RT	613058	13	124	2	8	1.3E-05	1	4	0	0
261	RT	613062	13	124	2	12	1.4E-06	1	4	0	0
262	RT	652958	14	95	2	8	6.6E-02	3	0	0	0
263	RT	652959	14	95	2	9	1.8E-03	3	0	0	1
264	RT	652960	14	95	2	10	4.2E-03	3	0	0	2
265	RT	652961	14	95	2	11	1.2E-04	3	0	0	2
266	RT	652962	14	95	2	12	7.5E-03	3	0	0	0
267	RT	652963	14	95	2	13	4.8E-04	3	0	0	2
268	RT	652964	14	95	2	14	2.3E-05	3	0	0	0
269	RT	652965	14	95	2	15	1.4E-06	3	0	0	2
270	RT	652973	14	95	3	8	8.6E-05	3	0	1	0
271	RT	652974	14	95	3	9	2.4E-06	3	0	1	1
272	RT	652975	14	95	3	10	3.5E-06	3	0	1	2
273	RT	652977	14	95	3	12	9.8E-06	3	0	1	0
274	RT	653243	14	96	2	8	1.1E-04	3	3	0	0
275	RT	653244	14	96	2	9	3.1E-06	3	3	0	1
276	RT	653245	14	96	2	10	7.2E-06	3	3	0	2
277	RT	653247	14	96	2	12	1.3E-05	3	3	0	0
278	RT	653528	14	97	2	8	6.7E-06	3	3	0	0
279	RT	653813	14	98	2	8	3.8E-04	3	4	0	0
280	RT	653814	14	98	2	9	1.0E-05	3	4	0	1
281	RT	653815	14	96	2	10	2.4E-05	3	4	0	2
282	RT	653817	14	98	2	12	4.3E-05	3	4	0	0
283	RT	653819	14	98	2	13	2.7E-06	3	4	0	2
284	RT	654098	14	99	2	8	9.4E-05	3	4	0	0
285	RT	654099	14	99	2	9	2.6E-06	3	4	0	1
286	RT	654100	14	99	2	10	6.0E-06	3	4	0	2
287	RT	654102	14	99	2	12	1.1E-05	3	4	0	0
288	RT	654383	14	100	2	8	2.0E-03	3	1	0	0
289	RT	654384	14	100	2	9	5.4E-05	3	1	0	1
290	RT	654385	14	100	2	10	1.3E-04	3	1	0	2
291	RT	654386	14	100	2	11	3.5E-06	3	1	0	2
292	RT	654387	14	100	2	12	2.2E-04	3	1	0	0
293	RT	654388	14	100	2	13	1.4E-05	3	1	0	2
294	RT	654398	14	100	3	8	2.6E-06	3	1	1	0

			--SYSTEM STATE BRANCHES----					----------CATEGORIES----------			
REC	INIT	SEQ#	SPC	FWS	ECC	PPC	EFREQ	SPC	FWS	ECC	PPC
295	RT	654668	14	101	2	8	3.4E-06	3	3	0	0
296	RT	655238	14	103	2	8	1.1E-05	3	4	0	0
297	RT	655242	14	103	2	12	1.3E-06	3	4	0	0
298	RT	655523	14	104	2	8	2.8E-06	3	4	0	0
299	RT	655808	14	105	2	8	8.7E-06	3	2	0	0
300	RT	657233	14	110	2	8	2.0E-04	3	1	0	0
301	RT	657234	14	110	2	9	5.4E-06	3	1	0	1
302	RT	657235	14	110	2	10	1.3E-05	3	1	0	2
303	RT	657237	14	110	2	12	2.2E-05	3	1	0	0
304	RT	657238	14	110	2	13	1.4E-06	3	1	0	2
305	RT	658088	14	113	2	8	1.1E-06	3	4	0	0
306	RT	660083	14	120	2	6	2.0E-03	3	0	0	0
307	RT	660084	14	120	2	9	5.4E-05	3	0	0	1
308	RT	660085	14	120	2	10	1.3E-04	3	0	0	2
309	RT	660086	14	120	2	11	3.5E-06	3	0	0	2
310	RT	660087	14	120	2	12	2.2E-04	3	0	0	0
311	RT	660088	14	120	2	13	1.4E-05	3	0	0	2
312	RT	660098	14	120	3	8	2.6E-06	3	0	1	0
313	RT	660368	14	121	2	8	3.4E-06	3	3	0	0
314	RT	660938	14	123	2	8	1.1E-05	3	4	0	0
315	RT	660942	14	123	2	12	1.3E-06	3	4	0	0
316	RT	661223	14	124	2	8	2.8E-06	3	4	0	0
317	RT	701123	15	95	2	6	2.7E-04	4	0	0	0
318	RT	701124	15	95	2	9	7.5E-06	4	0	0	1
319	RT	701125	15	95	2	10	1.7E-05	4	0	0	2
320	RT	701127	15	95	2	12	3.1E-05	4	0	0	0
321	RT	701128	15	95	2	13	2.0E-06	4	0	0	2
322	RT	701978	15	98	2	8	1.5E-06	4	4	0	0
323	RT	702548	15	100	2	6	8.1E-06	4	1	0	0
324	RT	708248	15	120	2	8	8.1E-06	4	0	0	0
325	RT	749288	16	95	2	6	6.0E-05	3	0	0	0
326	RT	749289	16	95	2	9	1.6E-06	3	0	0	1
327	RT	749290	16	95	2	10	3.8E-06	3	0	0	2
328	RT	749292	16	95	2	12	6.8E-06	3	0	0	0
329	RT	750713	16	100	2	8	1.8E-06	3	1	0	0
330	RT	756413	16	120	2	8	1.8E-06	3	0	0	0
331	RT	797453	17	95	2	8	2.7E-05	1	0	0	0
332	RT	797455	17	95	2	10	1.7E-06	1	0	0	2
333	RT	797457	17	95	2	12	3.1E-06	1	0	0	0
334	RT	845618	18	95	2	8	6.0E-06	3	0	0	0
335	RT	1134608	24	95	2	8	3.2E-04	1	0	0	0
336	RT	1134609	24	95	2	9	8.7E-06	1	0	0	1
337	RT	1134610	24	95	2	10	2.0E-05	1	0	0	2
338	RT	1134612	24	95	2	12	3.6E-05	1	0	0	0
339	RT	1134613	24	95	2	13	2.3E-06	1	0	0	2
340	RT	1135463	24	98	2	8	1.8E-06	1	4	0	0
341	RT	1136033	24	100	2	8	9.4E-06	1	1	0	0
342	RT	1136037	24	100	2	12	1.1E-06	1	1	0	0
343	RT	1141733	24	120	2	8	9.4E-06	1	0	0	0

			--SYSTEM STATE BRANCHES----					----------CATEGORIES----------			
REC	INIT	SEQ#	SPC	FWS	ECC	PPC	EFREQ	SPC	FWS	ECC	PPC
344	RT	1141737	24	120	2	12	1.1E-06	1	0	0	0
345	RT	1182773	25	95	2	8	3.6E-03	2	0	0	0
346	RT	1182774	25	95	2	9	9.8E-05	2	0	0	1
347	RT	1182775	25	95	2	10	2.3E-04	2	0	0	2
348	RT	1182776	25	95	2	11	6.2E-06	2	0	0	2
349	RT	1182777	25	95	2	12	4.0E-04	2	0	0	0
350	RT	1182778	25	95	2	13	2.6E-05	2	0	0	2
351	RT	1182779	25	95	2	14	1.2E-06	2	0	0	0
352	RT	1182788	25	95	3	8	4.6E-06	2	0	1	0
353	RT	1183058	25	96	2	8	6.1E-06	2	3	0	0
354	RT	1183628	25	98	2	8	2.0E-05	2	4	0	0
355	RT	1183630	25	98	2	10	1.3E-06	2	4	0	2
356	RT	1183632	25	98	2	12	2.3E-06	2	4	0	0
357	RT	1183913	25	99	2	8	5.0E-06	2	4	0	0
358	RT	1184198	25	100	2	8	1.1E-04	2	1	0	0
359	RT	1184199	25	100	2	9	2.9E-06	2	1	0	1
360	RT	1184200	25	100	2	10	6.7E-06	2	1	0	2
361	RT	1184202	25	100	2	12	1.2E-05	2	1	0	0
362	RT	1187048	25	110	2	8	1.1E-05	2	1	0	0
363	RT	1187052	25	110	2	12	1.3E-06	2	1	0	0
364	RT	1189898	25	120	2	8	1.1E-04	2	0	0	0
365	RT	1189899	25	120	2	9	2.9E-06	2	0	0	1
366	RT	1189900	25	120	2	10	5.8E-06	2	0	0	2
367	RT	1189902	25	120	2	12	1.2E-05	2	0	0	0
368	RT	1230938	26	95	2	8	2.7E-04	4	0	0	0
369	RT	1230939	26	95	2	9	7.5E-06	4	0	0	1
370	RT	1230940	26	95	2	10	1.7E-05	4	0	0	2
371	RT	1230942	26	95	2	12	3.1E-05	4	0	0	0
372	RT	1230943	26	95	2	13	2.0E-06	4	0	0	2
373	RT	1231793	26	98	2	8	1.5E-06	4	4	0	0
374	RT	1232363	26	100	2	8	8.1E-06	4	1	0	0
375	RT	1238063	26	120	2	8	8.1E-06	4	0	0	0
376	RT	1279103	27	95	2	8	3.9E-04	5	0	0	0
377	RT	1279104	27	95	2	9	1.1E-05	5	0	0	1
378	RT	1279105	27	95	2	10	2.5E-05	5	0	0	2
379	RT	1279107	27	95	2	12	4.5E-05	5	0	0	0
380	RT	1279108	27	95	2	13	2.9E-06	5	0	0	2
381	RT	1279958	27	98	2	8	2.2E-06	5	4	0	0
382	RT	1280529	27	100	2	8	1.2E-05	5	1	0	0
383	RT	1280532	27	100	2	12	1.3E-06	5	1	0	0
384	RT	1283378	27	110	2	8	1.2E-06	5	1	0	0
385	RT	1286228	27	120	2	8	1.2E-05	5	0	0	0
386	RT	1286232	27	120	2	12	1.3E-06	5	0	0	0
387	RT	1327268	28	95	2	8	3.0E-05	4	0	0	0
388	RT	1327270	28	95	2	10	1.9E-06	4	0	0	2
389	RT	1327272	28	95	2	12	3.4E-06	4	0	0	0
390	RT	1712588	36	95	2	8	3.6E-06	2	0	0	0
391	RT	1760753	37	95	2	8	2.7E-04	1	0	0	0
392	RT	1760754	37	95	2	9	7.5E-06	1	0	0	1

			--SYSTEM STATE BRANCHES----					----------CATEGORIES----------			
REC	INIT	SEQ#	SPC	FWS	ECC	PPC	EFREQ	SPC	FWS	ECC	PPC
491	EMFW	77828	2	105	2	8	2.1E-05	1	2	0	0
492	EMFW	77830	2	105	2	10	1.3E-06	1	2	0	2
493	EMFW	77832	2	105	2	12	2.4E-06	1	2	0	0
494	EMFW	79253	2	110	2	8	5.3E-04	1	1	0	0
495	EMFW	79254	2	110	2	9	1.5E-05	1	1	0	1
496	EMFW	79255	2	110	2	10	3.4E-05	1	1	0	2
497	EMFW	79257	2	110	2	12	6.1E-05	1	1	0	0
498	EMFW	79258	2	110	2	13	3.9E-06	1	1	0	2
499	EMFW	80108	2	113	2	8	3.0E-06	1	4	0	0
500	EMFW	124568	3	100	2	8	5.6E-05	2	1	0	0
501	EMFW	124569	3	100	2	9	1.6E-06	2	1	0	1
502	EMFW	124570	3	100	2	10	3.6E-06	2	1	0	2
503	EMFW	124572	3	100	2	12	6.4E-06	2	1	0	0
504	EMFW	127418	3	110	2	8	6.3E-06	2	1	0	0
505	EMFW	172733	4	100	2	8	4.3E-06	1	1	0	0
506	EMFW	199501	5	25	1	1	6.8E-06	0	1	0	0
507	EMFW	536656	12	25	1	1	7.0E-05	0	1	0	0
508	EMFW	538366	12	31	1	1	7.8E-06	0	1	0	0
509	EMFW	606218	13	100	2	8	4.8E-03	1	1	0	0
510	EMFW	606219	13	100	2	9	1.3E-04	1	1	0	1
511	EMFW	606220	13	100	2	10	3.1E-04	1	1	0	2
512	EMFW	606221	13	100	2	11	8.5E-06	1	1	0	2
513	EMFW	606222	13	100	2	12	5.4E-04	1	1	0	0
514	EMFW	606223	13	100	2	13	3.5E-05	1	1	0	2
515	EMFW	606224	13	100	2	14	1.6E-06	1	1	0	0
516	EMFW	606233	13	100	3	8	6.3E-06	1	1	1	0
517	EMFW	606503	13	101	2	8	8.2E-06	1	3	0	0
518	EMFW	607073	13	103	2	8	2.7E-05	1	4	0	0
519	EMFW	607075	13	103	2	10	1.7E-06	1	4	0	2
520	EMFW	607077	13	103	2	12	3.1E-06	1	4	0	0
521	EMFW	607358	13	104	2	8	6.8E-06	1	4	0	0
522	EMFW	607643	13	105	2	8	2.1E-05	1	2	0	0
523	EMFW	607645	13	105	2	10	1.4E-06	1	2	0	2
524	EMFW	607647	13	105	2	12	2.4E-06	1	2	0	0
525	EMFW	609068	13	110	2	8	5.4E-04	1	1	0	0
526	EMFW	609069	13	110	2	9	1.5E-05	1	1	0	1
527	EMFW	609070	13	110	2	10	3.4E-05	1	1	0	2
528	EMFW	609072	13	110	2	12	6.1E-05	1	1	0	0
529	EMFW	609073	13	110	2	13	3.9E-06	1	1	0	2
530	EMFW	609923	13	113	2	8	3.0E-06	1	4	0	0
531	EMFW	654383	14	100	2	8	1.1E-03	3	1	0	0
532	EMFW	654384	14	100	2	9	2.9E-05	3	1	0	1
533	EMFW	654385	14	100	2	10	6.7E-05	3	1	0	2
534	EMFW	654386	14	100	2	11	1.9E-06	3	1	0	2
535	EMFW	654387	14	100	2	12	1.2E-04	3	1	0	0
536	EMFW	654388	14	100	2	13	7.6E-06	3	1	0	2
537	EMFW	654398	14	100	3	8	1.4E-06	3	1	1	0
538	EMFW	654668	14	101	2	8	1.8E-06	3	3	0	0
539	EMFW	655238	14	103	2	8	6.0E-06	3	4	0	0

			--SYSTEM STATE BRANCHES----					----------CATEGORIES----------			
REC	INIT	SEQ#	SPC	FWS	ECC	PPC	EFREQ	SPC	FWS	ECC	PPC
442	RT	6914413	144	95	2	13	3.2E-06	6	0	0	2
443	RT	6915263	144	98	2	8	2.5E-06	6	4	0	0
444	RT	6915833	144	100	2	8	1.3E-05	6	1	0	0
445	RT	6915837	144	100	2	12	1.5E-06	6	1	0	0
446	RT	6918683	144	110	2	8	1.3E-06	6	1	0	0
447	RT	6921533	144	120	2	8	1.3E-05	6	0	0	0
448	RT	6921537	144	120	2	12	1.5E-06	6	0	0	0
449	EMFW	6841	1	25	1	1	7.6E-02	0	1	0	0
450	EMFW	7148	1	26	2	8	1.1E-04	0	3	0	0
451	EMFW	7149	1	26	2	9	2.9E-06	0	3	0	1
452	EMFW	7150	1	26	2	10	6.8E-06	0	3	0	2
453	EMFW	7152	1	26	2	12	1.2E-05	0	3	0	0
454	EMFW	7433	1	27	2	8	6.3E-06	0	3	0	0
455	EMFW	7703	1	28	1	8	3.5E-04	0	4	0	0
456	EMFW	7704	1	28	1	9	9.8E-06	0	4	0	1
457	EMFW	7705	1	28	1	10	2.3E-05	0	4	0	2
458	EMFW	7707	1	28	1	12	4.0E-05	0	4	0	0
459	EMFW	7708	1	28	1	13	2.6E-06	0	4	0	2
460	EMFW	7988	1	29	1	8	3.9E-05	0	4	0	0
461	EMFW	7989	1	29	1	9	2.4E-06	0	4	0	1
462	EMFW	7990	1	29	1	10	5.7E-06	0	4	0	2
463	EMFW	7992	1	29	1	12	1.0E-05	0	4	0	0
464	EMFW	8288	1	30	2	8	2.6E-04	0	2	0	0
465	EMFW	8289	1	30	2	9	7.7E-06	0	2	0	1
466	EMFW	8290	1	30	2	10	1.8E-05	0	2	0	2
467	EMFW	8292	1	30	2	12	3.2E-05	0	2	0	0
468	EMFW	8293	1	30	2	13	2.0E-06	0	2	0	2
469	EMFW	8551	1	31	1	1	8.4E-03	0	1	0	0
470	EMFW	8858	1	32	2	8	1.2E-05	0	3	0	0
471	EMFW	8862	1	32	2	12	1.3E-06	0	3	0	0
472	EMFW	9413	1	34	1	8	4.0E-05	0	4	0	0
473	EMFW	9414	1	34	1	9	1.1E-06	0	4	0	1
474	EMFW	9415	1	34	1	10	2.5E-06	0	4	0	2
475	EMFW	9417	1	34	1	12	4.5E-06	0	4	0	0
476	EMFW	9698	1	35	1	8	9.9E-06	0	4	0	0
477	EMFW	9702	1	35	1	12	1.1E-06	0	4	0	0
478	EMFW	76403	2	100	2	8	4.8E-03	1	1	0	0
479	EMFW	76404	2	100	2	9	1.3E-04	1	1	0	1
480	EMFW	76405	2	100	2	10	3.1E-04	1	1	0	2
481	EMFW	76406	2	100	2	11	8.4E-06	1	1	0	2
482	EMFW	76407	2	100	2	12	5.4E-04	1	1	0	0
483	EMFW	76408	2	100	2	13	3.5E-05	1	1	0	2
484	EMFW	76409	2	100	2	14	1.6E-06	1	1	0	0
485	EMFW	76418	2	100	3	8	6.2E-06	1	1	1	0
486	EMFW	76688	2	101	2	8	8.2E-06	1	3	0	0
487	EMFW	77258	2	103	2	8	2.7E-05	1	4	0	0
488	EMFW	77260	2	103	2	10	1.7E-06	1	4	0	2
489	EMFW	77262	2	103	2	12	3.1E-06	1	4	0	0
490	EMFW	77543	2	104	2	8	6.8E-06	1	4	0	0

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                         --SYSTEM STATE BRANCHES----        ----------CATEGORIES----------
REC   INIT     SEQ#     SPC    FWS    ECC    PPC   EFREQ      SPC     FWS     ECC     PPC

 540  EMFW    655523     14    104      2      8 1.5E-06       3       4       0       0
 541  EMFW    655808     14    105      2      8 4.7E-06       3       2       0       0
 542  EMFW    657233     14    110      2      8 1.2E-04       3       1       0       0
 543  EMFW    657234     14    110      2      9 3.2E-06       3       1       0       1
 544  EMFW    657235     14    110      2     10 7.5E-06       3       1       0       2
 545  EMFW    657237     14    110      2     12 1.3E-05       3       1       0       0
 546  EMFW    702548     15    100      2      8 4.3E-06       4       1       0       0
 547  EMFW   1136033     24    100      2      8 5.0E-06       1       1       0       0
 548  EMFW   1184198     25    100      2      8 5.6E-05       2       1       0       0
 549  EMFW   1184199     25    100      2      9 1.6E-06       2       1       0       1
 550  EMFW   1184200     25    100      2     10 3.6E-06       2       1       0       2
 551  EMFW   1184202     25    100      2     12 6.4E-06       2       1       0       0
 552  EMFW   1187048     25    110      2      8 6.3E-06       2       1       0       0
 553  EMFW   1232363     26    100      2      8 4.3E-06       4       1       0       0
 554  EMFW   1280528     27    100      2      8 6.3E-06       5       1       0       0
 555  EMFW   1762178     37    100      2      8 4.3E-06       1       1       0       0
 556  EMFW   2318761     49     25      1      1 6.8E-06       0       1       0       0
 557  EMFW   6364621    133     25      1      1 7.0E-05       0       1       0       0
 558  EMFW   6366331    133     31      1      1 7.9E-06       0       1       0       0
 559  EMFW   6434183    134    100      2      8 4.5E-06       1       1       0       0
 560  EMFW   6915833    144    100      2      8 7.0E-06       6       1       0       0
 561  LSLB   5786356    121     24      1      1 8.7E-05       0       0       0       0
 562  LSLB   5786641    121     25      1      1 2.6E-06       0       1       0       0
 563  LSLB   5790061    121     37      1      1 2.6E-06       0       0       0       0
 564  LSLB   5854778    122     95      2      8 5.5E-06       1       0       0       0
 565  LSLB   6075346    127     24      1      1 8.4E-04       0       0       0       0
 566  LSLB   6075631    127     25      1      1 2.5E-05       0       1       0       0
 567  LSLB   6077341    127     31      1      1 2.3E-06       0       1       0       0
 568  LSLB   6079051    127     37      1      1 2.5E-05       0       0       0       0
 569  SSLB   5786356    121     24      1      1 8.7E-05       0       0       0       0
 570  SSLB   5786641    121     25      1      1 2.6E-04       0       1       0       0
 571  SSLB   5787503    121     28      1      8 1.2E-06       0       4       0       0
 572  SSLB   5788351    121     31      1      1 2.6E-05       0       1       0       0
 573  SSLB   5790061    121     37      1      1 2.6E-04       0       0       0       0
 574  SSLB   5790923    121     40      1      8 1.2E-06       0       4       0       0
 575  SSLB   5834543    122     24      2      8 5.5E-04       1       0       0       0
 576  SSLB   5834544    122     24      2      9 1.5E-05       1       0       0       1
 577  SSLB   5834545    122     24      2     10 3.5E-05       1       0       0       2
 578  SSLB   5834547    122     24      2     12 5.2E-05       1       0       0       0
 579  SSLB   5834548    122     24      2     13 4.0E-06       1       0       0       2
 580  SSLB   5834828    122     25      2      8 1.6E-05       1       1       0       0
 581  SSLB   5834830    122     25      2     10 1.0E-06       1       1       0       2
 582  SSLB   5834832    122     25      2     12 1.8E-06       1       1       0       0
 583  SSLB   5836538    122     31      2      8 1.6E-06       1       1       0       0
 584  SSLB   5838248    122     37      2      8 1.6E-05       1       0       0       0
 585  SSLB   5838250    122     37      2     10 1.0E-06       1       0       0       2
 586  SSLB   5838252    122     37      2     12 1.8E-06       1       0       0       0
 587  SSLB   5882708    123     24      2      8 6.5E-06       2       0       0       0
 588  SSLB   6316171    132     24      1      1 8.0E-06       0       0       0       0
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REC	INIT	SEQ#	SPC	FWS	ECC	PPC	EFREQ	SPC	FWS	ECC	PPC
			--SYSTEM STATE BRANCHES----					----------CATEGORIES----------			
589	SSLB	6364336	133	24	1	1	7.5E-06	0	0	0	0
590	LOMFW	45323	1	160	1	8	4.9E-01	0	0	0	0
591	LOMFW	45324	1	160	1	9	1.3E-02	0	0	0	1
592	LOMFW	45325	1	160	1	10	3.1E-02	0	0	0	2
593	LOMFW	45326	1	160	1	11	8.6E-04	0	0	0	2
594	LOMFW	45327	1	160	1	12	5.5E-02	0	0	0	0
595	LOMFW	45328	1	160	1	13	3.5E-03	0	0	0	2
596	LOMFW	45329	1	160	1	14	1.7E-04	0	0	0	0
597	LOMFW	45330	1	160	1	15	1.1E-05	0	0	0	2
598	LOMFW	45608	1	161	1	8	1.7E-03	0	3	0	0
599	LOMFW	45609	1	161	1	9	4.6E-05	0	3	0	1
600	LOMFW	45610	1	161	1	10	1.1E-04	0	3	0	2
601	LOMFW	45611	1	161	1	11	2.9E-06	0	3	0	2
602	LOMFW	45612	1	161	1	12	1.9E-04	0	3	0	0
603	LOMFW	45613	1	161	1	13	1.2E-05	0	3	0	2
604	LOMFW	45893	1	162	1	8	1.7E-04	0	3	0	0
605	LOMFW	45894	1	162	1	9	4.6E-06	0	3	0	1
606	LOMFW	45895	1	162	1	10	1.1E-05	0	3	0	2
607	LOMFW	45897	1	162	1	12	1.9E-05	0	3	0	0
608	LOMFW	45898	1	162	1	13	1.2E-06	0	3	0	2
609	LOMFW	46178	1	163	1	8	2.8E-03	0	4	0	0
610	LOMFW	46179	1	163	1	9	7.6E-05	0	4	0	1
611	LOMFW	46180	1	163	1	10	1.8E-04	0	4	0	2
612	LOMFW	46181	1	163	1	11	4.9E-06	0	4	0	2
613	LOMFW	46182	1	163	1	12	3.1E-04	0	4	0	0
614	LOMFW	46183	1	163	1	13	2.0E-05	0	4	0	2
615	LOMFW	46463	1	164	1	8	6.9E-04	0	4	0	0
616	LOMFW	46464	1	164	1	9	1.9E-05	0	4	0	1
617	LOMFW	46465	1	164	1	10	4.4E-05	0	4	0	2
618	LOMFW	46466	1	164	1	11	1.2E-06	0	4	0	2
619	LOMFW	46467	1	164	1	12	7.8E-05	0	4	0	0
620	LOMFW	46468	1	164	1	13	5.0E-06	0	4	0	2
621	LOMFW	93503	2	160	2	8	3.7E-03	1	0	0	0
622	LOMFW	93504	2	160	2	9	1.0E-03	1	0	0	1
623	LOMFW	93505	2	160	2	10	2.4E-03	1	0	0	2
624	LOMFW	93506	2	160	2	11	6.6E-05	1	0	0	2
625	LOMFW	93507	2	160	2	12	4.2E-03	1	0	0	0
626	LOMFW	93508	2	160	2	13	2.7E-04	1	0	0	2
627	LOMFW	93509	2	160	2	14	1.3E-05	1	0	0	0
628	LOMFW	93518	2	160	3	8	4.9E-05	1	0	1	0
629	LOMFW	93519	2	160	3	9	1.3E-06	1	0	1	1
630	LOMFW	93520	2	160	3	10	3.1E-06	1	0	1	2
631	LOMFW	93522	2	160	3	12	5.5E-06	1	0	1	0
632	LOMFW	93788	2	161	2	8	1.3E-04	1	3	0	0
633	LOMFW	93789	2	161	2	9	3.5E-06	1	3	0	1
634	LOMFW	93790	2	161	2	10	8.1E-06	1	3	0	2
635	LOMFW	93792	2	161	2	12	1.4E-05	1	3	0	0
636	LOMFW	94073	2	162	2	8	1.3E-05	1	3	0	0
637	LOMFW	94077	2	162	2	12	1.4E-06	1	3	0	0

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			--SYSTEM STATE BRANCHES----					----------CATEGORIES----------			
REC	INIT	SEQ#	SPC	FWS	ECC	PPC	EFREQ	SPC	FWS	ECC	PPC
638	LOMFW	94358	2	163	2	8	2.1E-04	1	4	0	0
639	LOMFW	94359	2	163	2	9	5.8E-06	1	4	0	1
640	LOMFW	94360	2	163	2	10	1.3E-05	1	4	0	2
641	LOMFW	94362	2	163	2	12	2.4E-05	1	4	0	0
642	LOMFW	94363	2	163	2	13	1.5E-06	1	4	0	2
643	LOMFW	94643	2	164	2	8	5.3E-05	1	4	0	0
644	LOMFW	94644	2	164	2	9	1.5E-06	1	4	0	1
645	LOMFW	94645	2	164	2	10	3.4E-06	1	4	0	2
646	LOMFW	94647	2	164	2	12	6.0E-06	1	4	0	0
647	LOMFW	141668	3	160	2	8	4.4E-04	2	0	0	0
648	LOMFW	141669	3	160	2	9	1.2E-05	2	0	0	1
649	LOMFW	141670	3	160	2	10	2.8E-05	2	0	0	2
650	LOMFW	141672	3	160	2	12	5.0E-05	2	0	0	0
651	LOMFW	141673	3	160	2	13	3.2E-06	2	0	0	2
652	LOMFW	141953	3	161	2	8	1.5E-06	2	3	0	0
653	LOMFW	142523	3	163	2	8	2.5E-06	2	4	0	0
654	LOMFW	189833	4	160	2	8	3.4E-05	1	0	0	0
655	LOMFW	189835	4	160	2	10	2.1E-06	1	0	0	2
656	LOMFW	189837	4	160	2	12	3.8E-06	1	0	0	0
657	LOMFW	237983	5	160	1	8	4.4E-05	0	0	0	0
658	LOMFW	237984	5	160	1	9	1.3E-06	0	0	0	1
659	LOMFW	237985	5	160	1	10	2.8E-06	0	0	0	2
660	LOMFW	237987	5	160	1	12	5.0E-06	0	0	0	0
661	LOMFW	286163	6	160	2	8	3.4E-06	1	0	0	0
662	LOMFW	575138	12	160	1	8	4.3E-04	0	0	0	0
663	LOMFW	575139	12	160	1	9	1.3E-05	0	0	0	1
664	LOMFW	575140	12	160	1	10	2.9E-05	0	0	0	2
665	LOMFW	575142	12	160	1	12	5.1E-05	0	0	0	0
666	LOMFW	575143	12	160	1	13	3.3E-06	0	0	0	2
667	LOMFW	575423	12	161	1	8	1.5E-06	0	3	0	0
668	LOMFW	575993	12	163	1	8	2.6E-06	0	4	0	0
669	LOMFW	623318	13	160	2	8	3.7E-03	1	0	0	0
670	LOMFW	623319	13	160	2	9	1.0E-03	1	0	0	1
671	LOMFW	623320	13	160	2	10	2.4E-03	1	0	0	2
672	LOMFW	623321	13	160	2	11	6.6E-05	1	0	0	2
673	LOMFW	623322	13	160	2	12	4.3E-03	1	0	0	0
674	LOMFW	623323	13	160	2	13	2.7E-04	1	0	0	2
675	LOMFW	623324	13	160	2	14	1.3E-05	1	0	0	0
676	LOMFW	623333	13	160	3	8	4.9E-05	1	0	1	0
677	LOMFW	623334	13	160	3	9	1.3E-06	1	0	1	1
678	LOMFW	623335	13	160	3	10	3.1E-06	1	0	1	2
679	LOMFW	623337	13	160	3	12	5.5E-06	1	0	1	0
680	LOMFW	623603	13	161	2	8	1.3E-04	1	3	0	0
681	LOMFW	623604	13	161	2	9	3.5E-06	1	3	0	1
682	LOMFW	623605	13	161	2	10	8.2E-06	1	3	0	2
683	LOMFW	623607	13	161	2	12	1.4E-05	1	3	0	0
684	LOMFW	623888	13	162	2	8	1.3E-05	1	3	0	0
685	LOMFW	623892	13	162	2	12	1.4E-06	1	3	0	0
686	LOMFW	624173	13	163	2	8	2.1E-04	1	4	0	0

			--SYSTEM STATE BRANCHES----					----------CATEGORIES----------			
REC	INIT	SEQ#	SPC	FWS	ECC	PPC	EFREQ	SPC	FWS	ECC	PPC
687	LOMFW	624174	13	163	2	9	5.8E-06	1	4	0	1
688	LOMFW	624175	13	163	2	10	1.4E-05	1	4	0	2
689	LOMFW	624177	13	163	2	12	2.4E-05	1	4	0	0
690	LOMFW	624178	13	163	2	13	1.5E-06	1	4	0	2
691	LOMFW	624458	13	164	2	8	5.3E-05	1	4	0	0
692	LOMFW	624459	13	164	2	9	1.5E-06	1	4	0	1
693	LOMFW	624460	13	164	2	10	3.4E-06	1	4	0	2
694	LOMFW	624462	13	164	2	12	6.0E-06	1	4	0	0
695	LOMFW	671483	14	160	2	8	8.2E-03	3	0	0	0
696	LOMFW	671484	14	160	2	9	2.3E-04	3	0	0	1
697	LOMFW	671485	14	160	2	10	5.3E-04	3	0	0	2
698	LOMFW	671486	14	160	2	11	1.4E-05	3	0	0	2
699	LOMFW	671487	14	160	2	12	9.3E-04	3	0	0	0
700	LOMFW	671488	14	160	2	13	6.0E-05	3	0	0	2
701	LOMFW	671489	14	160	2	14	2.8E-06	3	0	0	0
702	LOMFW	671498	14	160	3	8	1.1E-05	3	0	1	0
703	LOMFW	671502	14	160	3	12	1.2E-06	3	0	1	0
704	LOMFW	671768	14	161	2	8	2.8E-05	3	3	0	0
705	LOMFW	671770	14	161	2	10	1.8E-06	3	3	0	2
706	LOMFW	671772	14	161	2	12	3.2E-06	3	3	0	0
707	LOMFW	672053	14	162	2	8	2.8E-06	3	3	0	0
708	LOMFW	672338	14	163	2	8	4.7E-05	3	4	0	0
709	LOMFW	672339	14	163	2	9	1.3E-06	3	4	0	1
710	LOMFW	672340	14	163	2	10	3.0E-06	3	4	0	2
711	LOMFW	672342	14	163	2	12	5.3E-06	3	4	0	0
712	LOMFW	672623	14	164	2	8	1.2E-05	3	4	0	0
713	LOMFW	672627	14	164	2	12	1.3E-06	3	4	0	0
714	LOMFW	719648	15	160	2	8	3.4E-05	4	0	0	0
715	LOMFW	719650	15	160	2	10	2.2E-06	4	0	0	2
716	LOMFW	719652	15	160	2	12	3.8E-06	4	0	0	0
717	LOMFW	767813	16	160	2	8	7.4E-06	3	0	0	0
718	LOMFW	815978	17	160	2	8	3.4E-06	1	0	0	0
719	LOMFW	1153133	24	160	2	8	3.9E-05	1	0	0	0
720	LOMFW	1153134	24	160	2	9	1.1E-06	1	0	0	1
721	LOMFW	1153135	24	160	2	10	2.5E-06	1	0	0	2
722	LOMFW	1153137	24	160	2	12	4.4E-06	1	0	0	0
723	LOMFW	1201298	25	160	2	8	4.4E-04	2	0	0	0
724	LOMFW	1201299	25	160	2	9	1.2E-05	2	0	0	1
725	LOMFW	1201300	25	160	2	10	2.8E-05	2	0	0	2
726	LOMFW	1201302	25	160	2	12	5.0E-05	2	0	0	0
727	LOMFW	1201303	25	160	2	13	3.2E-06	2	0	0	2
728	LOMFW	1201583	25	161	2	8	1.5E-06	2	3	0	0
729	LOMFW	1202153	25	163	2	8	2.5E-06	2	4	0	0
730	LOMFW	1249463	26	160	2	8	3.4E-05	4	0	0	0
731	LOMFW	1249465	26	160	2	10	2.1E-06	4	0	0	2
732	LOMFW	1249467	26	160	2	12	3.8E-06	4	0	0	0
733	LOMFW	1297628	27	160	2	8	4.9E-05	5	0	0	0
734	LOMFW	1297629	27	160	2	9	1.3E-06	5	0	0	1
735	LOMFW	1297630	27	160	2	10	3.1E-06	5	0	0	2

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REC	INIT	SEQ#	--SYSTEM STATE BRANCHES---- SPC	FWS	ECC	PPC	EFREQ	----------CATEGORIES---------- SPC	FWS	ECC	PPC
736	LOMFW	1297632	27	160	2	12	5.5E-06	5	0	0	0
737	LOMFW	1345793	28	160	2	8	3.7E-06	4	0	0	0
738	LOMFW	1779278	37	160	2	8	3.4E-05	1	0	0	0
739	LOMFW	1779280	37	160	2	10	2.2E-06	1	0	0	2
740	LOMFW	1779282	37	160	2	12	3.8E-06	1	0	0	0
741	LOMFW	1827443	38	160	2	8	7.4E-06	3	0	0	0
742	LOMFW	1875608	39	160	2	8	3.8E-06	4	0	0	0
743	LOMFW	2357243	49	160	1	8	4.4E-05	0	0	0	0
744	LOMFW	2357244	49	160	1	9	1.2E-06	0	0	0	1
745	LOMFW	2357245	49	160	1	10	2.8E-06	0	0	0	2
746	LOMFW	2357247	49	160	1	12	5.0E-06	0	0	0	0
747	LOMFW	2405423	50	160	2	8	3.4E-06	1	0	0	0
748	LOMFW	2549903	53	160	1	8	4.9E-06	0	0	0	0
749	LOMFW	2935238	61	160	2	8	3.4E-06	1	0	0	0
750	LOMFW	6403103	133	160	1	8	4.6E-04	0	0	0	0
751	LOMFW	6403104	133	160	1	9	1.3E-05	0	0	0	1
752	LOMFW	6403105	133	160	1	10	2.9E-05	0	0	0	2
753	LOMFW	6403107	133	160	1	12	5.2E-05	0	0	0	0
754	LOMFW	6403108	133	160	1	13	3.3E-06	0	0	0	2
755	LOMFW	6403388	133	161	1	8	1.6E-06	0	3	0	0
756	LOMFW	6403958	133	163	1	8	2.6E-06	0	4	0	0
757	LOMFW	6451283	134	160	2	8	3.5E-05	1	0	0	0
758	LOMFW	6451285	134	160	2	10	2.2E-06	1	0	0	2
759	LOMFW	6451287	134	160	2	12	3.9E-06	1	0	0	0
760	LOMFW	6932933	144	160	2	8	5.4E-05	5	0	0	0
761	LOMFW	6932934	144	160	2	9	1.5E-06	6	0	0	1
762	LOMFW	6932935	144	160	2	10	3.5E-06	6	0	0	2
763	LOMFW	6932937	144	160	2	12	6.2E-06	6	0	0	0
764	SBL1	6578	1	24	2	8	6.6E-02	0	0	0	0
765	SBL1	6579	1	24	2	9	1.8E-03	0	0	0	1
766	SBL1	6580	1	24	2	10	4.2E-03	0	0	0	2
767	SBL1	6581	1	24	2	11	1.2E-04	0	0	0	3
768	SBL1	6582	1	24	2	12	7.5E-03	0	0	0	0
769	SBL1	6583	1	24	2	13	4.8E-04	0	0	0	2
770	SBL1	6584	1	24	2	14	2.3E-05	0	0	0	0
771	SBL1	6585	1	24	2	15	1.4E-06	0	0	0	2
772	SBL1	6593	1	24	3	8	8.6E-05	0	0	1	0
773	SBL1	6594	1	24	3	9	2.4E-06	0	0	1	1
774	SBL1	6595	1	24	3	10	5.5E-06	0	0	1	2
775	SBL1	6597	1	24	3	12	9.8E-06	0	0	1	0
776	SBL1	6863	1	25	2	8	1.9E-03	0	1	0	0
777	SBL1	6864	1	25	2	9	5.4E-05	0	1	0	1
778	SBL1	6865	1	25	2	10	1.2E-04	0	1	0	2
779	SBL1	6866	1	25	2	11	3.4E-06	0	1	0	2
780	SBL1	6867	1	25	2	12	2.2E-04	0	1	0	0
781	SBL1	6868	1	25	2	13	1.4E-05	0	1	0	2
782	SBL1	6878	1	25	3	8	2.5E-06	0	1	1	0
783	SBL1	7148	1	26	2	8	3.3E-06	0	3	0	0
784	SBL1	7718	1	28	2	8	1.1E-05	0	4	0	0

			--SYSTEM STATE BRANCHES----					----------CATEGORIES----------			
REC	INIT	SEQ#	SPC	FWS	ECC	PPC	EFREQ	SPC	FWS	ECC	PPC
785	SBL1	7722	1	28	2	12	1.2E-06	0	4	0	0
786	SBL1	8003	1	29	2	8	2.8E-06	0	4	0	0
787	SBL1	8288	1	30	2	8	8.7E-06	0	2	0	0
788	SBL1	8573	1	31	2	8	2.0E-04	0	1	0	0
789	SBL1	8574	1	31	2	9	5.4E-06	0	1	0	1
790	SBL1	8575	1	31	2	10	1.2E-05	0	1	0	2
791	SBL1	8577	1	31	2	12	2.2E-05	0	1	0	0
792	SBL1	8578	1	31	2	13	1.4E-06	0	1	0	2
793	SBL1	9428	1	34	2	8	1.1E-06	0	4	0	0
794	SBL1	10283	1	37	2	8	2.0E-03	0	0	0	0
795	SBL1	10284	1	37	2	9	5.4E-05	0	0	0	1
796	SBL1	10285	1	37	2	10	1.2E-04	0	0	0	2
797	SBL1	10286	1	37	2	11	3.4E-06	0	0	0	2
798	SBL1	10287	1	37	2	12	2.2E-04	0	0	0	0
799	SBL1	10288	1	37	2	13	1.4E-05	0	0	0	2
800	SBL1	10298	1	37	3	8	2.5E-06	0	0	1	0
801	SBL1	10568	1	38	2	8	3.3E-06	0	3	0	0
802	SBL1	11138	1	40	2	8	1.1E-05	0	4	0	0
803	SBL1	11142	1	40	2	12	1.3E-06	0	4	0	0
804	SBL1	11423	1	41	2	8	2.8E-06	0	4	0	0
805	SBL1	74978	2	95	2	8	5.0E-03	1	0	0	0
806	SBL1	74979	2	95	2	9	1.4E-04	1	0	0	1
807	SBL1	74980	2	95	2	10	3.2E-04	1	0	0	2
808	SBL1	74981	2	95	2	11	8.8E-06	1	0	0	2
809	SBL1	74982	2	95	2	12	5.7E-04	1	0	0	0
810	SBL1	74983	2	95	2	13	3.6E-05	1	0	0	2
811	SBL1	74984	2	95	2	14	1.7E-06	1	0	0	0
812	SBL1	74993	2	95	3	8	6.5E-06	1	0	1	0
813	SBL1	75263	2	96	2	8	8.5E-06	1	3	0	0
814	SBL1	75833	2	98	2	8	2.8E-05	1	4	0	0
815	SBL1	75835	2	98	2	10	1.8E-06	1	4	0	2
816	SBL1	75837	2	98	2	12	3.2E-06	1	4	0	0
817	SBL1	76118	2	99	2	8	7.1E-06	1	4	0	0
818	SBL1	76403	2	100	2	8	1.5E-04	1	1	0	0
819	SBL1	76404	2	100	2	9	4.1E-06	1	1	0	1
820	SBL1	76405	2	100	2	10	9.5E-06	1	1	0	2
821	SBL1	76407	2	100	2	12	1.7E-05	1	1	0	0
822	SBL1	76408	2	100	2	13	1.1E-06	1	1	0	2
823	SBL1	79253	2	110	2	8	1.5E-05	1	1	0	0
824	SBL1	79257	2	110	2	12	1.7E-06	1	1	0	0
825	SBL1	82103	2	120	2	8	1.5E-04	1	0	0	0
826	SBL1	82104	2	120	2	9	4.1E-06	1	0	0	1
827	SBL1	82105	2	120	2	10	9.5E-06	1	0	0	2
828	SBL1	82107	2	120	2	12	1.7E-05	1	0	0	0
829	SBL1	82108	2	120	2	13	1.1E-06	1	0	0	2
830	SBL1	123143	3	95	2	8	5.9E-05	2	0	0	0
831	SBL1	123144	3	95	2	9	1.6E-06	2	0	0	1
832	SBL1	123145	3	95	2	10	3.8E-06	2	0	0	2
833	SBL1	123147	3	95	2	12	6.7E-06	2	0	0	0

			--SYSTEM STATE BRANCHES----					----------CATEGORIES-------- -			
REC	INIT	SEQ#	SPC	FWS	ECC	PPC	EFREQ	SPC	FWS	ECC	PPC
834	SBL1	124568	3	100	2	8	1.8E-06	2	1	0	0
835	SBL1	130268	3	120	2	8	1.8E-06	2	0	0	0
836	SBL1	171308	4	95	2	8	4.5E-06	1	0	0	0
837	SBL1	199238	5	24	2	8	6.0E-06	0	0	0	0
838	SBL1	536393	12	24	2	8	6.1E-05	0	0	0	0
839	SBL1	536394	12	24	2	9	1.7E-06	0	0	0	1
840	SBL1	536395	12	24	2	10	3.9E-06	0	0	0	2
841	SBL1	536397	12	24	2	12	6.9E-06	0	0	0	0
842	SBL1	536678	12	25	2	8	1.8E-06	0	1	0	0
843	SBL1	540098	12	37	2	8	1.8E-06	0	0	0	0
844	SBL1	604793	13	95	2	8	5.0E-03	1	0	0	0
845	SBL1	604794	13	95	2	9	1.4E-04	1	0	0	1
846	SBL1	604795	13	95	2	10	3.2E-04	1	0	0	2
847	SBL1	604796	13	95	2	11	8.9E-06	1	0	0	2
848	SBL1	604797	13	95	2	12	5.7E-04	1	0	0	0
849	SBL1	604798	13	95	2	13	3.6E-05	1	0	0	2
850	SBL1	604799	13	95	2	14	1.7E-06	1	0	0	0
851	SBL1	604808	13	95	3	8	6.6E-06	1	0	1	0
852	SBL1	605078	13	96	2	8	8.6E-06	1	3	0	0
853	SBL1	605648	13	98	2	8	2.8E-05	1	4	0	0
854	SBL1	605650	13	98	2	10	1.9E-06	1	4	0	2
855	SBL1	605652	13	98	2	12	3.2E-06	1	4	0	0
856	SBL1	605933	13	99	2	8	7.1E-06	1	4	0	0
857	SBL1	606218	13	100	2	8	1.5E-04	1	1	0	0
858	SBL1	606219	13	100	2	9	4.1E-06	1	1	0	1
859	SBL1	606220	13	100	2	10	9.5E-06	1	1	0	2
860	SBL1	606222	13	100	2	12	1.7E-05	1	1	0	0
861	SBL1	606223	13	100	2	13	1.1E-06	1	1	0	2
862	SBL1	609068	13	110	2	8	1.5E-05	1	1	0	0
863	SBL1	609072	13	110	2	12	1.7E-06	1	1	0	0
864	SBL1	611918	13	120	2	8	1.5E-04	1	0	0	0
865	SBL1	611919	13	120	2	9	4.1E-06	1	0	0	1
866	SBL1	611920	13	120	2	10	9.6E-06	1	0	0	2
867	SBL1	611922	13	120	2	12	1.7E-05	1	0	0	0
868	SBL1	611923	13	120	2	13	1.1E-06	1	0	0	2
869	SBL1	652958	14	95	2	8	1.1E-03	3	0	0	0
870	SBL1	652959	14	95	2	9	3.0E-05	3	0	0	1
871	SBL1	652960	14	95	2	10	7.1E-05	3	0	0	2
872	SBL1	652961	14	95	2	11	1.9E-06	3	0	0	2
873	SBL1	652962	14	95	2	12	1.3E-04	3	0	0	0
874	SBL1	652963	14	95	2	13	8.0E-06	3	0	0	2
875	SBL1	652973	14	95	3	8	1.4E-06	3	0	1	0
876	SBL1	653243	14	96	2	8	1.9E-06	3	3	0	0
877	SBL1	653813	14	98	2	8	6.3E-06	3	4	0	0
878	SBL1	654098	14	99	2	8	1.6E-06	3	4	0	0
879	SBL1	654383	14	100	2	8	3.3E-05	3	1	0	0
880	SBL1	654385	14	100	2	10	2.1E-06	3	1	0	2
881	SBL1	654387	14	100	2	12	3.7E-06	3	1	0	0
882	SBL1	657233	14	110	2	8	3.3E-06	3	1	0	0

			--SYSTEM STATE	BRANCHES----				----------CATEGORIES----------			
REC	INIT	SEQ#	SPC	FWS	ECC	PPC	EFREQ	SPC	FWS	ECC	PPC
883	SBL1	660083	14	120	2	8	3.3E-05	3	0	0	0
884	SBL1	660085	14	120	2	10	2.1E-06	3	0	0	2
885	SBL1	660087	14	120	2	12	3.7E-06	3	0	0	0
886	SBL1	701123	15	95	2	8	4.5E-06	4	0	0	0
887	SBL1	1134608	24	95	2	8	5.3E-06	1	0	0	0
888	SBL1	1182773	25	95	2	8	5.9E-05	2	0	0	0
889	SBL1	1182774	25	95	2	9	1.6E-06	2	0	0	1
890	SBL1	1182775	25	95	2	10	3.8E-06	2	0	0	2
891	SBL1	1182777	25	95	2	12	6.7E-06	2	0	0	0
892	SBL1	1184198	25	100	2	8	1.8E-06	2	1	0	0
893	SBL1	1189898	25	120	2	8	1.8E-06	2	0	0	0
894	SBL1	1230938	26	95	2	8	4.5E-06	4	0	0	0
895	SBL1	1279103	27	95	2	8	6.6E-06	5	0	0	0
896	SBL1	1760753	37	95	2	8	4.5E-06	1	0	0	0
897	SBL1	2318498	49	24	2	8	6.0E-06	0	0	0	0
898	SBL1	6364358	133	24	2	8	6.2E-05	0	0	0	0
899	SBL1	6364359	133	24	2	9	1.7E-06	0	0	0	1
900	SBL1	6364360	133	24	2	10	3.9E-06	0	0	0	2
901	SBL1	6364362	133	24	2	12	7.0E-06	0	0	0	0
902	SBL1	6364643	133	25	2	8	1.8E-06	0	1	0	0
903	SBL1	6368063	133	37	2	8	1.8E-06	0	0	0	0
904	SBL1	6432758	134	95	2	8	4.7E-06	1	0	0	0
905	SBL1	6914408	144	95	2	8	7.3E-06	5	0	0	0
906	SBL2	6571	1	24	2	1	8.0E-03	0	0	0	0
907	SBL2	6586	1	24	3	1	1.[illegible]E-05	0	0	1	0
908	SBL2	6856	1	25	2	1	2.[illegible]E-04	0	1	0	0
909	SBL2	7711	1	28	2	1	1.3E-06	0	4	0	0
910	SBL2	8281	1	30	2	1	1.0E-06	0	2	0	0
911	SBL2	8566	1	31	2	1	2.4E-05	0	1	0	0
912	SBL2	10276	1	37	2	1	2.4E-04	0	3	0	0
913	SBL2	11131	1	40	2	1	1.3E-06	0	4	0	0
914	SBL2	74971	2	95	2	1	6.1E-04	1	0	0	0
915	SBL2	75256	2	96	2	1	1.0E-06	1	3	0	0
916	SBL2	75826	2	98	2	1	3.4E-06	1	4	0	0
917	SBL2	76396	2	100	2	1	1.8E-05	1	1	0	0
918	SBL2	79246	2	110	2	1	1.8E-06	1	1	0	0
919	SBL2	82096	2	120	2	1	1.8E-05	1	0	0	0
920	SBL2	123136	3	95	2	1	7.1E-06	2	0	0	0
921	SBL2	536386	12	24	2	1	7.4E-06	0	0	0	0
922	SBL2	604786	13	95	2	1	6.1E-04	1	0	0	0
923	SBL2	605071	13	96	2	1	1.0E-06	1	3	0	0
924	SBL2	605641	13	98	2	1	3.4E-06	1	4	0	0
925	SBL2	606211	13	100	2	1	1.8E-05	1	1	0	0
926	SBL2	609061	13	110	2	1	1.8E-06	1	1	0	0
927	SBL2	611911	13	120	2	1	1.8E-05	1	0	0	0
928	SBL2	652951	14	95	2	1	1.3E-04	3	0	0	0
929	SBL2	654376	14	100	2	1	4.0E-06	3	1	0	0
930	SBL2	660076	14	120	2	1	4.0E-06	3	0	0	0
931	SBL2	1182766	25	95	2	1	7.1E-06	2	0	0	0

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			--SYSTEM STATE BRANCHES----					----------CATEGORIES----------			
REC	INIT	SEQ#	SPC	FWS	ECC	PPC	EFREQ	SPC	FWS	ECC	PPC
932	SBL2	6364351	133	24	2	1	7.5E-06	0	0	0	0
933	SGTR	6571	1	24	2	1	8.0E-03	0	0	0	0
934	SGTR	6586	1	24	3	1	1.0E-05	0	0	1	0
935	SGTR	6856	1	25	2	1	2.4E-04	0	1	0	0
936	SGTR	7711	1	28	2	1	1.3E-06	0	4	0	0
937	SGTR	8281	1	30	2	1	1.0E-06	0	2	0	0
938	SGTR	8566	1	31	2	1	2.4E-05	0	1	0	0
939	SGTR	10276	1	37	2	1	2.4E-04	0	0	0	0
940	SGTR	11131	1	40	2	1	1.3E-06	0	4	0	0
941	SGTR	74971	2	95	2	1	6.1E-04	1	0	0	0
942	SGTR	75256	2	96	2	1	1.0E-06	1	3	0	0
943	SGTR	75826	2	98	2	1	3.4E-06	1	4	0	0
944	SGTR	76396	2	100	2	1	1.8E-05	1	1	0	0
945	SGTR	79246	2	110	2	1	1.8E-06	1	1	0	0
946	SGTR	82096	2	120	2	1	1.8E-05	1	0	0	0
947	SGTR	123136	3	95	2	1	7.1E-06	2	0	0	0
948	SGTR	536386	12	24	2	1	7.4E-06	0	0	0	0
949	SGTR	604786	13	95	2	1	6.1E-04	1	0	0	0
950	SGTR	605071	13	96	2	1	1.0E-06	1	3	0	0
951	SGTR	605641	13	98	2	1	3.4E-06	1	4	0	0
952	SGTR	606211	13	100	2	1	1.8E-05	1	1	0	0
953	SGTR	609061	13	110	2	1	1.8E-06	1	1	0	0
954	SGTR	611911	13	120	2	1	1.8E-05	1	0	0	0
955	SGTR	652951	14	95	2	1	1.3E-04	3	0	0	0
956	SGTR	654376	14	100	2	1	4.0E-06	3	1	0	0
957	SGTR	660076	14	120	2	1	4.0E-06	3	0	0	0
958	SGTR	1182766	25	95	2	1	7.1E-06	2	0	0	0
959	SGTR	6364351	133	24	2	1	7.5E-06	0	0	0	0
960	SI	6578	1	24	2	8	6.6E-03	0	0	0	0
961	SI	6579	1	24	2	9	1.8E-04	0	0	0	1
962	SI	6580	1	24	2	10	4.2E-04	0	0	0	2
963	SI	6581	1	24	2	11	1.2E-05	0	0	0	2
964	SI	6582	1	24	2	12	7.5E-04	0	0	0	0
965	SI	6583	1	24	2	13	4.8E-05	0	0	0	2
966	SI	6584	1	24	2	14	2.3E-06	0	0	0	0
967	SI	6863	1	25	2	8	2.0E-04	0	1	0	0
968	SI	6864	1	25	2	9	5.4E-06	0	1	0	1
969	SI	6865	1	25	2	10	1.2E-05	0	1	0	2
970	SI	6867	1	25	2	12	2.2E-05	0	1	0	0
971	SI	6868	1	25	2	13	1.4E-06	0	1	0	2
972	SI	7718	1	28	2	8	1.1E-06	0	4	0	0
973	SI	8573	1	31	2	8	2.0E-05	0	1	0	0
974	SI	8575	1	31	2	10	1.2E-06	0	1	0	2
975	SI	8577	1	31	2	12	2.2E-06	0	1	0	0
976	SI	10283	1	37	2	8	2.0E-04	0	0	0	0
977	SI	10284	1	37	2	9	5.4E-06	0	0	0	1
978	SI	10285	1	37	2	10	1.3E-05	0	0	0	2
979	SI	10287	1	37	2	12	2.2E-05	0	0	0	0
980	SI	10288	1	37	2	13	1.4E-06	0	0	0	2

			--SYSTEM STATE BRANCHES----					----------CATEGORIES----------			
REC	INIT	SEQ#	SPC	FWS	ECC	PPC	EFREQ	SPC	FWS	ECC	PPC
981	SI	11138	1	40	2	8	1.1E-06	0	4	0	0
982	SI	74978	2	95	2	8	5.0E-04	1	0	0	0
983	SI	74979	2	95	2	9	1.4E-05	1	0	0	1
984	SI	74980	2	95	2	10	3.2E-05	1	0	0	2
985	SI	74982	2	95	2	12	5.7E-05	1	0	0	0
986	SI	74983	2	95	2	13	3.6E-06	1	0	0	2
987	SI	75833	2	98	2	8	2.8E-06	1	4	0	0
988	SI	76403	2	100	2	8	1.5E-05	1	1	0	0
989	SI	76407	2	100	2	12	1.7E-06	1	1	0	0
990	SI	79253	2	110	2	8	1.5E-06	1	1	0	0
991	SI	82103	2	120	2	8	1.5E-05	1	0	0	0
992	SI	82107	2	120	2	12	1.7E-06	1	0	0	0
993	SI	123143	3	95	2	8	5.9E-06	2	0	0	0
994	SI	536393	12	24	2	8	6.1E-06	0	0	0	0
995	SI	604793	13	95	2	8	5.0E-04	1	0	0	0
996	SI	604794	13	95	2	9	1.4E-05	1	0	0	1
997	SI	604795	13	95	2	10	3.2E-05	1	0	0	2
998	SI	604797	13	95	2	12	5.7E-05	1	0	0	0
999	SI	604798	13	95	2	13	3.7E-06	1	0	0	2
1000	SI	605648	13	98	2	8	2.9E-06	1	4	0	0
1001	SI	606218	13	100	2	8	1.5E-05	1	1	0	[illegible]
1002	SI	606222	13	100	2	12	1.7E-06	1	1	0	0
1003	SI	609068	13	110	2	8	1.5E-06	1	1	0	0
1004	SI	611918	13	120	2	8	1.5E-05	1	0	0	0
1005	SI	611922	13	120	2	12	1.7E-06	1	0	0	0
1006	SI	652958	14	95	2	8	1.1E-04	3	0	0	0
1007	SI	652959	14	95	2	9	3.1E-06	3	0	0	1
1008	SI	652960	14	95	2	10	7.1E-06	3	0	0	2
1009	SI	652962	14	95	2	12	1.3E-05	3	0	0	0
1010	SI	654383	14	100	2	8	3.3E-06	3	1	0	0
1011	SI	660083	14	120	2	8	3.2E-06	3	0	0	0
1012	SI	1182773	25	95	2	8	5.9E-06	2	0	0	0
1013	SI	6364358	133	24	2	8	6.2E-06	0	0	0	0

A.2 Sequences Summed by System State Category

			SYSTEM STATE CATEGORIES				
REC	TYPE	INIT	SPC	FWS	ECC	PPC	EFREQ
1	NORM	RT	0	0	0	0	5.0E+00
2	FW	RT	0	1	0	0	1.6E-01
3	FW	RT	0	3	0	1	4.9E-04
4	FW	RT	0	3	0	2	3.0E-05
5	FW	RT	0	4	0	0	1.9E-03
6	FW	RT	0	4	0	1	4.7E-05
7	FW	RT	0	4	0	2	1.3E-04
8	FW	RT	0	2	0	0	5.8E-04
9	FW	RT	0	2	0	1	1.4E-05
10	FW	RT	0	2	0	2	3.7E-05
11	SL	RT	1	0	0	0	6.9E-01
12	SL	RT	1	0	0	1	1.7E-02
13	SL	RT	1	0	0	2	4.5E-02
14	SL	RT	1	0	1	0	9.0E-04
15	SL	RT	1	0	1	1	2.2E-05
16	SL	RT	1	0	1	2	5.6E-05
17	SS	RT	1	3	0	1	2.8E-05
18	SS	RT	1	3	0	2	7.7E-05
19	SS	RT	1	4	0	0	5.0E-03
20	SS	RT	1	4	0	1	1.2E-04
21	SS	RT	1	4	0	2	3.2E-04
22	SS	RT	1	4	1	0	4.4E-06
23	SL	RT	1	1	0	0	2.3E-02
24	SS	RT	1	1	0	1	5.4E-04
25	SS	RT	1	1	0	2	1.4E-03
26	SS	RT	1	1	1	0	2.8E-05
27	SS	RT	1	2	0	0	8.8E-05
28	SS	RT	1	2	0	1	2.2E-06
29	SS	RT	1	2	0	2	5.0E-06
30	SL	RT	2	0	0	0	8.2E-03
31	SL	RT	2	0	0	1	2.0E-04
32	SL	RT	2	0	0	2	5.3E-04
33	SL	RT	2	0	1	0	9.2E-06
34	SS	RT	2	3	0	0	1.3E-05
35	SS	RT	2	4	0	0	5.5E-05
36	SS	RT	2	4	0	2	2.6E-06
37	SS	RT	2	1	0	0	2.6E-04
38	SS	RT	2	1	0	1	5.8E-06
39	SS	RT	2	1	0	2	1.3E-05
40	SL	RT	3	0	0	0	7.6E-02
41	SL	RT	3	0	0	1	1.9E-03
42	SL	RT	3	0	0	2	5.0E-03
43	SL	RT	3	0	1	0	9.9E-05
44	SL	RT	3	0	1	1	2.4E-06
45	SL	RT	3	0	1	2	5.5E-06
46	SS	RT	3	3	0	0	1.4E-04
47	SS	RT	3	3	0	1	3.1E-06
48	SS	RT	3	3	0	2	7.2E-06
49	SS	RT	3	4	0	0	5.5E-04
50	SS	RT	3	4	0	1	1.3E-05
51	SS	RT	3	4	0	2	3.3E-05

BEST COPY AVAILABLE

SYSTEM STATE CATEGORIES

REC	TYPE	INIT	SPC	FWS	ECC	PPC	EFREQ
52	SS	RT	3	1	0	0	2.4E-03
53	SS	RT	3	1	0	1	6.0E-05
54	SS	RT	3	1	0	2	1.6E-04
55	SS	RT	3	1	1	0	2.6E-06
56	SS	RT	3	2	0	0	8.7E-06
57	SL	RT	4	0	0	0	6.9E-04
58	SL	RT	4	0	0	1	1.5E-05
59	SL	RT	4	0	0	2	4.3E-05
60	SS	RT	4	4	0	0	3.1E-06
61	SS	RT	4	1	0	0	1.6E-05
62	SL	RT	5	0	0	0	4.5E-04
63	SL	RT	5	0	0	1	1.1E-05
64	SL	RT	5	0	0	2	2.8E-05
65	SS	RT	5	4	0	0	2.2E-06
66	SS	RT	5	1	0	0	1.4E-05
67	SL	RT	6	0	0	0	5.0E-04
68	SL	RT	6	0	0	1	1.2E-05
69	SL	RT	6	0	0	2	3.1E-05
70	SS	RT	6	4	0	0	2.5E-06
71	SS	RT	6	1	0	0	1.6E-05
72	RES	RT	99	99	99	99	1.9E-04
73	FW	EMFW	0	1	0	0	8.4E-02
74	FW	EMFW	0	3	0	0	1.4E-04
75	FW	EMFW	0	3	0	1	2.9E-06
76	FW	EMFW	0	3	0	2	6.8E-06
77	FW	EMFW	0	4	0	0	5.5E-04
78	FW	EMFW	0	4	0	1	1.3E-05
79	FW	EMFW	0	4	0	2	3.3E-05
80	FW	EMFW	0	2	0	0	3.1E-04
81	FW	EMFW	0	2	0	1	7.7E-06
82	FW	EMFW	0	2	0	2	2.0E-05
83	SS	EMFW	1	1	0	0	1.2E-02
84	SS	EMFW	1	1	0	1	2.9E-04
85	SS	EMFW	1	1	0	2	7.7E-04
86	SS	EMFW	1	1	1	0	1.2E-05
87	SS	EMFW	1	3	0	0	1.6E-05
88	SS	EMFW	1	4	0	0	8.0E-05
89	SS	EMFW	1	4	0	2	3.5E-06
90	SS	EMFW	1	2	0	0	4.7E-05
91	SS	EMFW	1	2	0	2	2.7E-06
92	SS	EMFW	2	1	0	0	1.4E-04
93	SS	EMFW	2	1	0	1	3.1E-06
94	SS	EMFW	2	1	0	2	7.2E-06
95	SS	EMFW	3	1	0	0	1.3E-03
96	SS	EMFW	3	1	0	1	3.2E-05
97	SS	EMFW	3	1	0	2	8.4E-05
98	SS	EMFW	3	1	1	0	1.4E-06
99	SS	EMFW	3	3	0	0	1.8E-06
100	SS	EMFW	3	4	0	0	7.5E-06
101	SS	EMFW	3	2	0	0	4.7E-06
102	SS	EMFW	4	1	0	0	8.6E-06

SYSTEM STATE CATEGORIES

REC	TYPE	INIT	SPC	FWS	ECC	PPC	EFREQ
103	SS	EMFW	5	1	0	0	6.3E-06
104	SS	EMFW	6	1	0	0	7.0E-06
105	RES	EMFW	99	99	99	99	7.2E-05
106	SL	LSLB	1	0	0	0	9.6E-04
107	SL	LSLB	1	1	0	0	3.0E-05
108	SL	LSLB	3	0	0	0	5.5E-06
109	RES	LSLB	99	99	99	99	5.3E-06
110	SL	SSLB	1	0	0	0	9.0E-03
111	SL	SSLB	1	1	0	0	2.8E-04
112	SS	SSLB	1	4	0	0	2.4E-06
113	SL	SSLB	3	0	0	0	6.3E-04
114	SL	SSLB	3	0	0	1	1.5E-05
115	SL	SSLB	3	0	0	2	4.0E-05
116	SL	SSLB	3	1	0	0	2.0E-05
117	SL	SSLB	3	1	0	2	1.0E-06
118	SL	SSLB	4	0	0	0	6.5E-06
119	RES	SSLB	99	99	99	99	1.4E-05
120	NORM	LOMFW	0	0	0	0	5.5E-01
121	LOCA	LOMFW	0	0	0	1	1.4E-02
122	LOCA	LOMFW	0	0	0	2	3.6E-02
123	FW	LOMFW	0	3	0	0	2.0E-03
124	FW	LOMFW	0	3	0	1	5.1E-05
125	FW	LOMFW	0	3	0	2	1.3E-04
126	FW	LOMFW	0	4	0	0	3.9E-03
127	FW	LOMFW	0	4	0	1	9.5E-05
128	FW	LOMFW	0	4	0	2	2.5E-04
129	SL	LOMFW	1	0	0	0	8.4E-02
130	SL	LOMFW	1	0	0	1	2.1E-03
131	SL	LOMFW	1	0	0	2	5.5E-03
132	SL	LOMFW	1	0	1	0	1.1E-04
133	SL	LOMFW	1	0	1	1	2.7E-06
134	SL	LOMFW	1	0	1	2	6.2E-06
135	SS	LOMFW	1	3	0	0	3.1E-04
136	SS	LOMFW	1	3	0	1	7.0E-06
137	SS	LOMFW	1	3	0	2	1.6E-05
138	SS	LOMFW	1	4	0	0	5.9E-04
139	SS	LOMFW	1	4	0	1	1.5E-05
140	SS	LOMFW	1	4	0	2	3.7E-05
141	SL	LOMFW	2	0	0	0	9.8E-04
142	SL	LOMFW	2	0	0	1	2.4E-05
143	SL	LOMFW	2	0	0	2	6.3E-05
144	SS	LOMFW	2	3	0	0	3.0E-06
145	SS	LOMFW	2	4	0	0	5.0E-06
146	SL	LOMFW	3	0	0	0	9.2E-03
147	SL	LOMFW	3	0	0	1	2.3E-04
148	SL	LOMFW	3	0	0	2	6.0E-04
149	SL	LOMFW	3	0	1	0	1.2E-05
150	SS	LOMFW	3	3	0	0	3.4E-05
151	SS	LOMFW	3	3	0	2	1.8E-06
152	SS	LOMFW	3	4	0	0	6.5E-05
153	SS	LOMFW	3	4	0	1	1.3E-06

SYSTEM STATE CATEGORIES

REC	TYPE	INIT	SPC	FWS	ECC	PPC	EFREQ
154	SS	LOMFW	3	4	0	2	3.0E-06
155	SL	LOMFW	4	0	0	0	8.3E-05
156	SL	LOMFW	4	0	0	2	4.3E-06
157	SL	LOMFW	5	0	0	0	5.4E-05
158	SL	LOMFW	5	0	0	1	1.3E-06
159	SL	LOMFW	5	0	0	2	3.1E-06
160	SL	LOMFW	6	0	0	0	6.1E-05
161	SL	LOMFW	6	0	0	1	1.5E-06
162	SL	LOMFW	6	0	0	2	3.5E-06
163	RES	LOMFW	99	99	99	99	5.9E-05
164	LOCA	SBL1	0	0	0	0	7.6E-02
165	LOCA	SBL1	0	0	0	1	1.9E-03
166	LOCA	SBL1	0	0	0	2	5.0E-03
167	LOCA	SBL1	0	0	1	0	9.9E-05
168	LOCA	SBL1	0	0	1	1	2.4E-06
169	LOCA	SBL1	0	0	1	2	5.5E-06
170	LOCA	SBL1	0	1	0	0	2.4E-03
171	LOCA	SBL1	0	1	0	1	5.9E-05
172	LOCA	SBL1	0	1	0	2	1.6E-04
173	LOCA	SBL1	0	1	1	0	2.5E-06
174	LOCA	SBL1	0	3	0	0	6.6E-06
175	LOCA	SBL1	0	4	0	0	3.1E-05
176	LOCA	SBL1	0	2	0	0	8.7E-06
177	LOCA	SBL1	1	0	0	0	1.2E-02
178	LOCA	SBL1	1	0	0	1	2.9E-04
179	LOCA	SBL1	1	0	0	2	7.5E-04
180	LOCA	SBL1	1	0	1	0	1.3E-05
181	LOCA	SBL1	1	3	0	0	1.7E-05
182	LOCA	SBL1	1	4	0	0	7.7E-05
183	LOCA	SBL1	1	4	0	2	3.6E-06
184	LOCA	SBL1	1	1	0	0	3.7E-04
185	LOCA	SBL1	1	1	0	1	8.2E-06
186	LOCA	SBL1	1	1	0	2	2.1E-05
187	LOCA	SBL1	2	0	0	0	1.4E-04
188	LOCA	SBL1	2	0	0	2	7.6E-06
189	LOCA	SBL1	2	1	0	0	3.5E-06
190	LOCA	SBL1	3	0	0	0	1.3E-03
191	LOCA	SBL1	3	0	0	1	3.0E-05
192	LOCA	SBL1	3	0	0	2	8.3E-05
193	LOCA	SBL1	3	0	1	0	1.4E-06
194	LOCA	SBL1	3	3	0	0	1.9E-06
195	LOCA	SBL1	3	4	0	0	7.8E-06
196	LOCA	SBL1	3	1	0	0	4.0E-05
197	LOCA	SBL1	3	1	0	2	2.1E-06
198	LOCA	SBL1	4	0	0	0	9.1E-06
199	LOCA	SBL1	5	0	0	0	6.6E-06
200	LOCA	SBL1	6	0	0	0	7.3E-06
201	RES	SBL1	99	99	99	99	7.8E-05
202	LOCA	SBL2	0	0	0	0	8.2E-03
203	LOCA	SBL2	0	0	1	0	1.0E-05
204	LOCA	SBL2	0	1	0	0	2.6E-04

```
                      SYSTEM STATE CATEGORIES
REC TYPE INIT  SPC  FWS  ECC  PPC   EFREQ

205 LOCA SBL2    0    4    0    0  2.7E-06
206 LOCA SBL2    0    2    0    0  1.0E-06
207 LOCA SBL2    1    0    0    0  1.2E-03
208 LOCA SBL2    1    3    0    0  2.1E-06
209 LOCA SBL2    1    4    0    0  6.9E-06
210 LOCA SBL2    1    1    0    0  4.0E-05
211 LOCA SBL2    2    0    0    0  1.4E-05
212 LOCA SBL2    3    0    0    0  1.4E-04
213 LOCA SBL2    3    1    0    0  4.0E-06
214  RES SBL2   99   99   99   99  1.8E-05
215 SGTR SGTR    0    0    0    0  8.2E-03
216 SGTR SGTR    0    0    1    0  1.0E-05
217 SGTR SGTR    0    1    0    0  2.6E-04
218 SGTR SGTR    0    4    0    0  2.7E-06
219 SGTR SGTR    0    2    0    0  1.0E-06
220 SGTR SGTR    1    0    0    0  1.2E-03
221 SGTR SGTR    1    3    0    0  2.1E-06
222 SGTR SGTR    1    4    0    0  6.9E-06
223 SGTR SGTR    1    1    0    0  4.0E-05
224 SGTR SGTR    2    0    0    0  1.4E-05
225 SGTR SGTR    3    0    0    0  1.4E-04
226 SGTR SGTR    3    1    0    0  4.0E-06
227  RES SGTR   99   99   99   99  1.8E-05
228 NORM  SI     0    0    0    0  7.6E-03
229 LOCA  SI     0    0    0    1  1.9E-04
230 LOCA  SI     0    0    0    2  5.0E-04
231  FW   SI     0    1    0    0  2.4E-04
232 LOCA  SI     0    1    0    1  5.4E-06
233 LOCA  SI     0    1    0    2  1.5E-05
234  FW   SI     0    4    0    0  2.2E-06
235  SL   SI     1    0    0    0  1.2E-03
236  SL   SI     1    0    0    1  2.3E-05
237  SL   SI     1    0    0    2  7.2E-05
238  SS   SI     1    4    0    0  5.7E-06
239  SS   SI     1    1    0    0  3.6E-05
240  SL   SI     2    0    0    0  1.2E-05
241  SL   SI     3    0    0    0  1.3E-04
242  SL   SI     3    0    0    1  3.1E-06
243  SL   SI     3    0    0    2  7.1E-06
244  SS   SI     3    1    0    0  3.3E-06
245  RES  SI    99   99   99   99  3.9E-05
```

A.3 General Table of Vessel Through-Wall Crack Frequency Results

It should be noted that a review of the assignment of representative fracture-mechanics cases to each sequence revealed both conservative and non-conservative assignments. It was the opinion of the analysts that in general conservative and sometimes overly conservative assignments were made. This was particularly true for the assignment of a fracture mechanics case to the residual cases. An attempt was made to reconstruct the table and thus reduce the level of conservatism involved in this portion of the analysis. Unfortunately, there was little that could be done without performing additional thermal-hydraulic and/or fracture-mechanics calculations. In addition the resolution of the few non-conservative assignments led to a higher overall vessel failure value. Thus in trying to reduce both the conservative and non-conservative biases introduced by the assignment of fracture-mechanics cases, we produced a set of data which was clearly biased in the conservative direction. This data can be found in Appendix F and is presented since it includes the elimination of some of the non-conservative biases found in the analysis even though it fails to address the larger category of conservative biases.

```
REC TYPE INIT  SPC  FWS  ECC  PPC   EFREQ     TRANS      HF     TRANS#  TFREQ    P(F:T)   P(TWO)
  1 NORM  RT    0    0    0    0  5.0E+00    NONE    1.0E+00      0  5.0E+00  0.0E+00  0.0E+00
  2  FW   RT    0    1    0    0  1.6E-01  OVRFD3    1.0E+00     18  1.6E-01  1.0E-07  1.6E-08
  3  FW   RT    0    3    0    1  4.9E-04   PSBG5    9.9E-01     15  4.9E-04  1.0E-07  4.9E-11
  4  FW   RT    0    3    0    1  4.9E-04   PSBG4    1.0E-02     14  4.9E-06  1.3E-03  6.2E-09
  5  FW   RT    0    3    0    2  3.0E-05   PSBG5    9.9E-01     15  3.0E-05  1.0E-07  3.0E-12
  6  FW   RT    0    3    0    2  3.0E-05   PSBG4    1.0E-02     14  3.0E-07  1.3E-03  3.3E-10
  7  FW   RT    0    4    0    0  1.9E-03    NONE    9.9E-01      0  1.9E-03  0.0E+00  0.0E+00
  8  FW   RT    0    4    0    0  1.9E-03  OVRFD6    1.0E-02     21  1.9E-05  1.4E-06  2.7E-11
  9  FW   RT    0    4    0    0  1.9E-03  OVRFD7    3.6E-03     22  6.8E-06  1.0E-07  6.8E-13
 10  FW   RT    0    4    0    1  4.7E-05    NONE    9.9E-01      0  4.7E-05  0.0E+00  0.0E+00
 11  FW   RT    0    4    0    1  4.7E-05  OVRFD6    1.0E-02     21  4.7E-07  1.4E-06  6.6E-13
 12  FW   RT    0    4    0    1  4.7E-05  OVRFD7    3.6E-03     22  1.7E-07  1.0E-07  1.7E-14
 13  FW   RT    0    4    0    2  1.3E-04    NONE    9.9E-01      0  1.3E-04  0.0E+00  0.0E+00
 14  FW   RT    0    4    0    2  1.3E-04  OVRFD6    1.0E-02     21  1.3E-06  1.4E-06  1.8E-12
 15  FW   RT    0    4    0    2  1.3E-04  OVRFD7    3.6E-03     22  4.7E-07  1.0E-07  4.7E-14
 16  FW   RT    0    2    0    0  5.8E-04     MIN    9.9E-01     38  5.7E-04  1.0E-07  5.7E-11
 17  FW   RT    0    2    0    0  5.8E-04   INEL6    1.0E-02     56  5.8E-06  4.2E-07  2.4E-12
 18  FW   RT    0    2    0    1  1.4E-05   TBVG2    9.9E-01      2  1.4E-05  1.0E-07  1.4E-12
 19  FW   RT    0    2    0    1  1.4E-05   TBVG6    1.0E-02      6  1.4E-07  1.7E-06  2.4E-13
 20  FW   RT    0    2    0    2  3.7E-05   TBVG2    9.9E-01      2  3.7E-05  1.0E-07  3.7E-12
 21  FW   RT    0    2    0    2  3.7E-05   TBVG6    1.0E-02      6  3.7E-07  1.7E-06  6.3E-13
 22  SL   RT    1    0    0    0  6.9E-01   TBVG2    9.9E-01      2  6.8E-01  1.0E-07  6.8E-08
 23  SL   RT    1    0    0    0  6.9E-01   TBVG1    1.0E-02      1  6.9E-03  4.0E-07  2.8E-09
 24  SL   RT    1    0    0    1  1.7E-02   PSBG5    9.9E-01     15  1.7E-02  1.0E-07  1.7E-09
 25  SL   RT    1    0    0    1  1.7E-02   TBVG6    1.0E-02      6  1.7E-04  1.7E-06  2.9E-10
 26  SL   RT    1    0    0    2  4.5E-02   PSBG5    9.9E-01     15  4.5E-02  1.0E-07  4.5E-09
 27  SL   RT    1    0    0    2  4.5E-02   TBVG6    1.0E-02      6  4.5E-04  1.7E-06  7.7E-10
 28  SL   RT    1    0    1    0  9.0E-04   TBVG2    9.9E-01      2  8.9E-04  1.0E-07  8.9E-11
 29  SL   RT    1    0    1    0  9.0E-04   TBVG1    1.0E-02      1  9.0E-06  4.0E-07  3.6E-12
 30  SL   RT    1    0    1    1  2.2E-05   PSBG5    9.9E-01     15  2.2E-05  1.0E-07  2.2E-12
 31  SL   RT    1    0    1    1  2.2E-05   TBVG6    1.0E-02      6  2.2E-07  1.7E-06  3.7E-13
 32  SL   RT    1    0    1    2  5.6E-05   PSBG5    9.9E-01     15  5.5E-05  1.0E-07  5.5E-12
 33  SL   RT    1    0    1    2  5.6E-05   TBVG6    1.0E-02      6  5.6E-07  1.7E-06  9.5E-13
 34  SS   RT    1    3    0    1  2.8E-05   PSBG5    9.9E-01     15  2.8E-05  1.0E-07  2.8E-12
 35  SS   RT    1    3    0    1  2.8E-05   TBVG6    1.0E-02      6  2.8E-07  1.7E-06  4.8E-13
 36  SS   RT    1    3    0    2  7.7E-05   PSBG5    9.9E-01     15  7.6E-05  1.0E-07  7.6E-12
 37  SS   RT    1    3    0    2  7.7E-05   TBVG6    1.0E-02      6  7.7E-07  1.7E-06  1.3E-12
 38  SS   RT    1    4    0    0  5.0E-03   TBVG2    9.9E-01      2  5.0E-03  1.0E-07  5.0E-10
 39  SS   RT    1    4    0    0  5.0E-03  OVRFD6    1.0E-02     21  5.0E-05  1.4E-06  7.0E-11
 40  SS   RT    1    4    0    0  5.0E-03  OVRFD7    3.6E-03     22  1.8E-05  1.0E-07  1.8E-12
 41  SS   RT    1    4    0    1  1.2E-04   PSBG5    9.9E-01     15  1.2E-04  1.0E-07  1.2E-11
 42  SS   RT    1    4    0    1  1.2E-04  OVRFD6    1.0E-02     21  1.2E-06  1.4E-06  1.7E-12
 43  SS   RT    1    4    0    1  1.2E-04  OVRFD7    3.6E-03     22  4.3E-07  1.0E-07  4.3E-14
 44  SS   RT    1    4    0    2  3.2E-04   PSBG5    9.9E-01     15  3.2E-04  1.0E-07  3.2E-11
 45  SS   RT    1    4    0    2  3.2E-04  OVRFD6    1.0E-02     21  3.2E-06  1.4E-06  4.5E-12
 46  SS   RT    1    4    0    2  3.2E-04  OVRFD7    3.6E-03     22  1.2E-06  1.0E-07  1.2E-13
 47  SS   RT    1    4    1    0  4.4E-06   TBVG2    9.9E-01      2  4.4E-06  1.0E-07  4.4E-13
 48  SS   RT    1    4    1    0  4.4E-06  OVRFD6    1.0E-02     21  4.4E-08  1.4E-06  6.2E-14
 49  SS   RT    1    4    1    0  4.4E-06  OVRFD7    3.6E-03     22  1.6E-08  1.0E-07  1.6E-15
 50  SL   RT    1    1    0    0  2.2E-02   TBVG3    9.9E-01      3  2.2E-02  1.0E-07  2.2E-09
 51  SL   RT    1    1    0    0  2.2E-02   TBVG4    1.0E-02      4  2.2E-04  2.0E-03  4.4E-07
 52  SS   RT    1    1    0    1  5.4E-04   PSBG5    9.9E-01     15  5.3E-04  1.0E-07  5.3E-11
 53  SS   RT    1    1    0    1  5.4E-04   TBVG6    1.0E-02      6  5.4E-06  1.7E-06  9.2E-12
```

REC	TYPE	INIT	SPC	FWS	ECC	PPC	EFREQ	TRANS	HF	TRANS#	TFREQ	P(FIT)	P(TWO)
54	SS	RT	1	1	0	2	1.4E-03	PSBG5	9.9E-01	15	1.4E-03	1.0E-07	1.4E-10
55	SS	RT	1	1	0	2	1.4E-03	TBVG6	1.0E-02	6	1.4E-05	1.7E-06	2.4E-11
56	SS	RT	1	1	1	0	2.8E-05	TBVG3	9.9E-01	2	2.8E-05	1.0E-07	2.8E-12
57	SS	RT	1	1	1	0	2.8E-05	TBVG4	1.0E-02	4	2.8E-07	2.0E-03	5.6E-10
58	SS	RT	1	2	0	0	8.8E-05	TBVG3	9.9E-01	3	8.7E-05	1.0E-07	8.7E-12
59	SS	RT	1	2	0	0	8.8E-05	TBVG4	1.0E-02	4	8.8E-07	2.0E-03	1.8E-09
60	SS	RT	1	2	0	1	2.2E-06	PSBG5	9.9E-01	15	2.2E-06	1.0E-07	2.2E-13
61	SS	RT	1	2	0	1	2.2E-06	TBVG6	1.0E-02	6	2.2E-08	1.7E-06	3.7E-14
62	SS	RT	1	2	0	2	5.0E-06	PSBG5	9.9E-01	15	5.0E-06	1.0E-07	5.0E-13
63	SS	RT	1	2	0	2	5.0E-06	TBVG6	1.0E-02	6	5.0E-08	1.7E-06	8.5E-14
64	SL	RT	2	0	0	0	8.2E-03	TBVG2	9.9E-01	2	8.1E-03	1.0E-07	8.1E-10
65	SL	RT	2	0	0	0	8.2E-03	TBVG1	1.0E-02	1	8.2E-05	4.0E-07	3.3E-11
66	SL	RT	2	0	0	1	2.0E-04	PSBG5	9.9E-01	15	2.0E-04	1.0E-07	2.0E-11
67	SL	RT	2	0	0	1	2.0E-04	TBVG6	1.0E-02	6	2.0E-06	1.7E-06	3.4E-12
68	SL	RT	2	0	0	2	5.3E-04	PSBG5	9.9E-01	15	5.2E-04	1.0E-07	5.2E-11
69	SL	RT	2	0	0	2	5.3E-04	TBVG6	1.0E-02	6	5.3E-06	1.7E-06	9.0E-12
70	SL	RT	2	0	1	0	9.2E-06	TBVG2	9.9E-01	2	9.1E-06	1.0E-07	9.1E-13
71	SL	RT	2	0	1	0	9.2E-06	TBVG1	1.0E-02	1	9.2E-08	4.0E-07	3.7E-14
72	SS	RT	2	3	0	0	1.2E-05	TBVG2	9.9E-01	2	1.2E-05	1.0E-07	1.2E-12
73	SS	RT	2	3	0	0	1.2E-05	LANL7	1.0E-02	41	1.2E-07	1.8E-03	2.2E-10
74	SS	RT	2	4	0	0	5.5E-05	TBVG2	9.9E-01	2	5.4E-05	1.0E-07	5.4E-12
75	SS	RT	2	4	0	0	5.5E-05	OVRFD6	1.0E-02	21	5.5E-07	1.4E-06	7.7E-13
76	SS	RT	2	4	0	0	5.5E-05	OVRFD7	3.6E-03	22	2.0E-07	1.0E-07	2.0E-14
77	SS	RT	2	4	0	2	2.6E-06	PSBG5	9.9E-01	15	2.6E-06	1.0E-07	2.6E-13
78	SS	RT	2	4	0	2	2.6E-06	OVRFD6	1.0E-02	21	2.6E-08	1.4E-06	3.6E-14
79	SS	RT	2	4	0	2	2.6E-06	OVRFD7	3.6E-03	22	9.4E-09	1.0E-07	9.4E-16
80	SS	RT	2	1	0	0	2.6E-04	TBVG3	9.9E-01	3	2.6E-04	1.0E-07	2.6E-11
81	SS	RT	2	1	0	0	2.6E-04	TBVG4	1.0E-02	4	2.6E-06	2.0E-03	5.2E-09
82	SS	RT	2	1	0	1	5.8E-06	PSBG5	9.9E-01	15	5.7E-06	1.0E-07	5.7E-13
83	SS	RT	2	1	0	1	5.8E-06	TBVG6	1.0E-02	6	5.8E-08	1.7E-06	9.9E-14
84	SS	RT	2	1	0	2	1.3E-05	PSBG5	9.9E-01	15	1.3E-05	1.0E-07	1.3E-12
85	SS	RT	2	1	0	2	1.3E-05	TBVG6	1.0E-02	6	1.3E-07	1.7E-06	2.2E-13
86	SL	RT	3	0	0	0	7.6E-02	TBVG8	9.9E-01	8	7.5E-02	1.0E-07	7.5E-09
87	SL	RT	3	0	0	0	7.6E-02	TBVG5	1.0E-02	5	7.6E-04	5.2E-06	4.0E-09
88	SL	RT	3	0	0	1	1.9E-03	TBVG8	9.9E-01	8	1.9E-03	1.0E-07	1.9E-10
89	SL	RT	3	0	0	1	1.9E-03	TBVG10	1.0E-02	10	1.9E-05	1.0E-05	1.9E-10
90	SL	RT	3	0	0	2	5.0E-03	TBVG8	9.9E-01	8	5.0E-03	1.0E-07	5.0E-10
91	SL	RT	3	0	0	2	5.0E-03	TBVG10	1.0E-02	10	5.0E-05	1.0E-05	5.0E-10
92	SL	RT	3	0	1	0	9.9E-05	TBVG8	9.9E-01	8	9.8E-05	1.0E-07	9.8E-12
93	SL	RT	3	0	1	0	9.9E-05	TBVG5	1.0E-02	5	9.9E-07	5.2E-06	5.1E-12
94	SL	RT	3	0	1	1	2.4E-06	TBVG8	9.9E-01	8	2.4E-06	1.0E-07	2.4E-13
95	SL	RT	3	0	1	1	2.4E-06	TBVG10	1.0E-02	10	2.4E-08	1.0E-05	2.4E-13
96	SL	RT	3	0	1	2	5.5E-06	TBVG8	9.9E-01	8	5.4E-06	1.0E-07	5.4E-13
97	SL	RT	3	0	1	2	5.5E-06	TBVG10	1.0E-02	10	5.5E-08	1.0E-05	5.5E-13
98	SS	RT	3	3	0	0	1.4E-04	TBVG2	9.9E-01	2	1.4E-04	1.0E-07	1.4E-11
99	SS	RT	3	3	0	0	1.4E-04	LANL7	1.0E-02	41	1.4E-06	1.8E-03	2.5E-09
100	SS	RT	3	3	0	1	3.1E-06	PSBG5	9.9E-01	15	3.1E-06	1.0E-07	3.1E-13
101	SS	RT	3	3	0	1	3.1E-06	TBVG6	1.0E-02	6	3.1E-08	1.7E-06	5.3E-14
102	SS	RT	3	3	0	2	7.2E-06	PSBG5	9.9E-01	15	7.1E-06	1.0E-07	7.1E-13
103	SS	RT	3	3	0	2	7.2E-06	TBVG6	1.0E-02	6	7.2E-08	1.7E-06	1.2E-13
104	SS	RT	3	4	0	0	5.5E-04	TBVG2	9.9E-01	2	5.4E-04	1.0E-07	5.4E-11
105	SS	RT	3	4	0	0	5.5E-04	OVRFD6	1.0E-02	21	5.5E-06	1.4E-06	7.7E-12
106	SS	RT	3	4	0	0	5.5E-04	OVRFD7	3.6E-03	22	2.0E-06	1.0E-07	2.0E-13

BEST

REC	TYPE	INIT	SPC	FWS	ECC	PPC	EFREQ	TRANS	HF	TRANS#	TFREQ	P(FIT)	P(TWC)
107	SS	RT	3	4	0	1	1.3E-05	PSBG5	9.9E-01	15	1.3E-05	1.0E-07	1.3E-12
108	SS	RT	3	4	0	1	1.3E-05	OVRFD6	1.0E-02	21	1.3E-07	1.4E-06	1.8E-13
109	SS	RT	3	4	0	1	1.3E-05	OVRFD7	3.6E-03	22	4.7E-08	1.0E-07	4.7E-15
110	SS	RT	3	4	0	2	3.3E-05	PSBG5	9.9E-01	15	3.3E-05	1.0E-07	3.3E-12
111	SS	RT	3	4	0	2	3.3E-05	OVRFD6	1.0E-02	21	3.3E-07	1.4E-06	4.6E-13
112	SS	RT	3	4	0	2	3.3E-05	OVRFD7	3.6E-03	22	1.2E-07	1.0E-07	1.2E-14
113	SS	RT	3	1	0	0	2.4E-03	TBVG8	9.9E-01	8	2.4E-03	1.0E-07	2.4E-10
114	SS	RT	3	1	0	0	2.4E-03	TBVG9	1.0E-02	9	2.4E-05	2.0E-03	4.8E-08
115	SS	RT	3	1	0	1	6.0E-05	TBVG8	9.9E-01	8	5.9E-05	1.0E-07	5.9E-12
116	SS	RT	3	1	0	1	6.0E-05	TBVG10	1.0E-02	10	6.0E-07	1.0E-05	6.0E-12
117	SS	RT	3	1	0	2	1.6E-04	TBVG8	9.9E-01	8	1.6E-04	1.0E-07	1.6E-11
118	SS	RT	3	1	0	2	1.6E-04	TBVG10	1.0E-02	10	1.6E-06	1.0E-05	1.6E-11
119	SS	RT	3	1	1	0	2.6E-06	TBVG8	9.9E-01	8	2.6E-06	1.0E-07	2.6E-13
120	SS	RT	3	1	1	0	2.6E-06	TBVG5	1.0E-02	5	2.6E-08	5.2E-06	1.4E-13
121	SS	RT	3	2	0	0	8.7E-06	TBVG3	9.9E-01	3	8.6E-06	1.0E-07	8.6E-13
122	SS	RT	3	2	0	0	8.7E-06	TBVG4	1.0E-02	4	8.7E-08	2.0E-03	1.7E-10
123	SL	RT	4	0	0	0	6.9E-04	TBVG2	9.9E-01	2	6.8E-04	1.0E-07	6.8E-11
124	SL	RT	4	0	0	0	6.9E-04	TBVG1	1.0E-02	1	6.9E-06	4.0E-07	2.8E-12
125	SL	RT	4	0	0	1	1.5E-05	TBVG2	9.9E-01	2	1.5E-05	1.0E-07	1.5E-12
126	SL	RT	4	0	0	1	1.5E-05	TBVG10	1.0E-02	10	1.5E-07	1.0E-05	1.5E-12
127	SL	RT	4	0	0	2	4.3E-05	TBVG2	9.9E-01	2	4.3E-05	1.0E-07	4.3E-12
128	SL	RT	4	0	0	2	4.3E-05	TBVG10	1.0E-02	10	4.3E-07	1.0E-05	4.3E-12
129	SS	RT	4	4	0	0	3.1E-06	TBVG2	9.9E-01	2	3.1E-06	1.0E-07	3.1E-13
130	SS	RT	4	4	0	0	3.1E-06	OVRFD6	1.0E-02	21	3.1E-08	1.4E-06	4.3E-14
131	SS	RT	4	4	0	0	3.1E-06	OVRFD7	3.6E-03	22	1.1E-08	1.0E-07	1.1E-15
132	SS	RT	4	1	0	0	1.6E-05	TBVG2	9.9E-01	2	1.6E-05	1.0E-07	1.6E-12
133	SS	RT	4	1	0	0	1.6E-05	TBVG9	1.0E-02	9	1.6E-07	2.0E-03	3.2E-10
134	SL	RT	5	0	0	0	4.5E-04	TBVG2	9.9E-01	2	4.5E-04	1.0E-07	4.5E-11
135	SL	RT	5	0	0	0	4.5E-04	TBVG5	1.0E-02	5	4.5E-06	5.2E-06	2.3E-11
136	SL	RT	5	0	0	1	1.1E-05	TBVG2	9.9E-01	2	1.1E-05	1.0E-07	1.1E-12
137	SL	RT	5	0	0	1	1.1E-05	TBVG10	1.0E-02	10	1.1E-07	1.0E-05	1.1E-12
138	SL	RT	5	0	0	2	2.8E-05	TBVG2	9.9E-01	2	2.8E-05	1.0E-07	2.8E-12
139	SL	RT	5	0	0	2	2.8E-05	TBVG10	1.0E-02	10	2.8E-07	1.0E-05	2.8E-12
140	SS	RT	5	4	0	0	2.2E-06	TBVG2	9.9E-01	2	2.2E-06	1.0E-07	2.2E-13
141	SS	RT	5	4	0	0	2.2E-06	OVRFD6	1.0E-02	21	2.2E-08	1.4E-06	3.1E-14
142	SS	RT	5	4	0	0	2.2E-06	OVRFD7	3.6E-03	22	7.9E-09	1.0E-07	7.9E-16
143	SS	RT	5	1	0	0	1.4E-05	TBVG2	9.9E-01	2	1.4E-05	1.0E-07	1.4E-12
144	SS	RT	5	1	0	0	1.4E-05	TBVG9	1.0E-02	9	1.4E-07	2.0E-03	2.8E-10
145	SL	RT	6	0	0	0	5.0E-04	MSLB1	9.9E-01	26	5.0E-04	6.2E-04	3.1E-07
146	SL	RT	6	0	0	0	5.0E-04	TBVG9	1.0E-02	9	5.0E-06	2.0E-03	1.0E-08
147	SL	RT	6	0	0	1	1.2E-05	MSLB7	9.9E-01	32	1.2E-05	6.2E-04	7.4E-09
148	SL	RT	6	0	0	1	1.2E-05	TBVG9	1.0E-02	9	1.2E-07	2.0E-03	2.4E-10
149	SL	RT	6	0	0	2	3.1E-05	MSLB7	9.9E-01	32	3.1E-05	6.2E-04	1.9E-08
150	SL	RT	6	0	0	2	3.1E-05	TBVG9	1.0E-02	9	3.1E-07	2.0E-03	6.2E-10
151	SS	RT	6	4	0	0	2.5E-06	MSLB1	9.9E-01	26	2.5E-06	6.2E-04	1.5E-09
152	SS	RT	6	4	0	0	2.5E-06	OVRFD6	1.0E-02	21	2.5E-08	1.4E-06	3.5E-14
153	SS	RT	6	4	0	0	2.5E-06	OVRFD7	3.6E-03	22	9.0E-09	1.0E-07	9.0E-16
154	SS	RT	6	1	0	0	1.6E-05	MSLB7	9.9E-01	32	1.6E-05	6.2E-04	9.9E-09
155	SS	RT	6	1	0	0	1.6E-05	TBVG9	1.0E-02	9	1.6E-07	2.0E-03	3.2E-10
156	RES	RT	99	99	99	99	1.9E-04	DEF	1.0E+00	99	1.9E-04	5.4E-03	1.0E-06
157	FW	EMFW	0	1	0	0	8.4E-02	OVRFD3	1.0E+00	18	8.4E-02	1.0E-07	8.4E-09
158	FW	EMFW	0	3	0	0	1.4E-04	MIN	9.9E-01	98	1.4E-04	1.0E-07	1.4E-11
159	FW	EMFW	0	3	0	0	1.4E-04	INEL3	1.0E-02	53	1.4E-06	1.0E-07	1.4E-13

BEST COPY AVA

REC	TYPE	INIT	SPC	FWS	ECC	FPC	EFREQ	TRANS	HF	TRANS#	TFREQ	P(FIT)	P(TWC)
160	FW	EMFW	0	3	0	1	2.9E-06	PSBG5	9.9E-01	15	2.9E-06	1.0E-07	2.9E-13
161	FW	EMFW	0	3	0	1	2.9E-06	PSBG4	1.0E-02	14	2.9E-08	1.3E-03	3.7E-11
162	FW	EMFW	0	3	0	2	6.8E-06	PSBG5	9.9E-01	15	6.7E-06	1.0E-07	6.7E-13
163	FW	EMFW	0	3	0	2	6.8E-06	PSBG4	1.0E-02	14	6.8E-08	1.3E-03	8.6E-11
164	FW	EMFW	0	4	0	0	5.5E-04	TBVG2	9.9E-01	2	5.4E-04	1.0E-07	5.4E-11
165	FW	EMFW	0	4	0	0	5.5E-04	OVRFD6	1.0E-02	21	5.5E-06	1.4E-06	7.7E-12
166	FW	EMFW	0	4	0	0	5.5E-04	OVRFD7	3.6E-03	22	2.0E-06	1.0E-07	2.0E-13
167	FW	EMFW	0	4	0	1	1.3E-05	PSBG5	9.9E-01	15	1.3E-05	1.0E-07	1.3E-12
168	FW	EMFW	0	4	0	1	1.3E-05	OVRFD6	1.0E-02	21	1.3E-07	1.4E-06	1.8E-13
169	FW	EMFW	0	4	0	1	1.3E-05	OVRFD7	3.6E-03	22	4.7E-08	1.0E-07	4.7E-15
170	FW	EMFW	0	4	0	2	3.3E-05	PSBG5	9.9E-01	15	3.3E-05	1.0E-07	3.3E-12
171	FW	EMFW	0	4	0	2	3.3E-05	OVRFD6	1.0E-02	21	3.3E-07	1.4E-06	4.6E-13
172	FW	EMFW	0	4	0	2	3.3E-05	OVRFD7	3.6E-03	22	1.2E-07	1.0E-07	1.2E-14
173	FW	EMFW	0	2	0	0	3.1E-04	TBVG2	9.9E-01	2	3.1E-04	1.0E-07	3.1E-11
174	FW	EMFW	0	2	0	0	3.1E-04	TBVG1	1.0E-02	1	3.1E-06	4.0E-07	1.2E-12
175	FW	EMFW	0	2	0	1	7.7E-06	TBVG2	9.9E-01	2	7.6E-06	1.0E-07	7.6E-13
176	FW	EMFW	0	2	0	1	7.7E-06	TBVG6	1.0E-02	6	7.7E-08	1.7E-06	1.3E-13
177	FW	EMFW	0	2	0	2	2.0E-05	TBVG2	9.9E-01	2	2.0E-05	1.0E-07	2.0E-12
178	FW	EMFW	0	2	0	2	2.0E-05	TBVG6	1.0E-02	6	2.0E-07	1.7E-06	3.4E-13
179	SS	EMFW	1	1	0	0	1.2E-02	TBVG2	9.9E-01	2	1.2E-02	1.0E-07	1.2E-09
180	SS	EMFW	1	1	0	0	1.2E-02	TBVG1	1.0E-02	1	1.2E-04	4.0E-07	4.8E-11
181	SS	EMFW	1	1	0	1	2.9E-04	PSBG5	9.9E-01	15	2.9E-04	1.0E-07	2.9E-11
182	SS	EMFW	1	1	0	1	2.9E-04	TBVG6	1.0E-02	6	2.9E-06	1.7E-06	4.9E-12
183	SS	EMFW	1	1	0	2	7.7E-04	PSBG5	9.9E-01	15	7.6E-04	1.0E-07	7.6E-11
184	SS	EMFW	1	1	0	2	7.7E-04	TBVG6	1.0E-02	6	7.7E-06	1.7E-06	1.3E-11
185	SS	EMFW	1	1	1	0	1.2E-05	TBVG2	9.9E-01	2	1.2E-05	1.0E-07	1.2E-12
186	SS	EMFW	1	1	1	0	1.2E-05	TBVG1	1.0E-02	1	1.2E-07	4.0E-07	4.8E-14
187	SS	EMFW	1	3	0	0	1.6E-05	TBVG2	9.9E-01	2	1.6E-05	1.0E-07	1.6E-12
188	SS	EMFW	1	3	0	0	1.6E-05	LANL7	1.0E-02	41	1.6E-07	1.8E-03	2.9E-10
189	SS	EMFW	1	4	0	0	8.0E-05	TBVG2	9.9E-01	2	7.9E-05	1.0E-07	7.9E-12
190	SS	EMFW	1	4	0	0	8.0E-05	OVRFD6	1.0E-02	21	8.0E-07	1.4E-06	1.1E-12
191	SS	EMFW	1	4	0	0	8.0E-05	OVRFD7	3.6E-03	22	2.9E-07	1.0E-07	2.9E-14
192	SS	EMFW	1	4	0	2	3.5E-06	PSBG5	9.9E-01	15	3.5E-06	1.0E-07	3.5E-13
193	SS	EMFW	1	4	0	2	3.5E-06	OVRFD6	1.0E-02	21	3.5E-08	1.4E-06	4.9E-14
194	SS	EMFW	1	4	0	2	3.5E-06	OVRFD7	3.6E-03	22	1.3E-08	1.0E-07	1.3E-15
195	SS	EMFW	1	2	0	0	4.7E-05	TBVG3	9.9E-01	3	4.7E-05	1.0E-07	4.7E-12
196	SS	EMFW	1	2	0	0	4.7E-05	TBVG4	1.0E-02	4	4.7E-07	2.0E-03	9.4E-10
197	SS	EMFW	1	2	0	2	2.7E-06	PSBG5	9.9E-01	15	2.7E-06	1.0E-07	2.7E-13
198	SS	EMFW	1	2	0	2	2.7E-06	TBVG6	1.0E-02	6	2.7E-08	1.7E-06	4.6E-14
199	SS	EMFW	2	1	0	0	1.4E-04	TBVG3	9.9E-01	3	1.4E-04	1.0E-07	1.4E-11
200	SS	EMFW	2	1	0	0	1.4E-04	TBVG4	1.0E-02	4	1.4E-06	2.0E-03	2.8E-09
201	SS	EMFW	2	1	0	1	3.1E-06	PSBG5	9.9E-01	15	3.1E-06	1.0E-07	3.1E-13
202	SS	EMFW	2	1	0	1	3.1E-06	TBVG6	1.0E-02	6	3.1E-08	1.7E-06	5.3E-14
203	SS	EMFW	2	1	0	2	7.2E-06	PSBG5	9.9E-01	15	7.1E-06	1.0E-07	7.1E-13
204	SS	EMFW	2	1	0	2	7.2E-06	TBVG6	1.0E-02	6	7.2E-08	1.7E-06	1.2E-13
205	SS	EMFW	3	1	0	0	1.3E-03	TBVG8	9.9E-01	8	1.3E-03	1.0E-07	1.3E-10
206	SS	EMFW	3	1	0	0	1.3E-03	TBVG9	1.0E-02	9	1.3E-05	2.0E-03	2.6E-08
207	SS	EMFW	3	1	0	1	3.2E-05	TBVG8	9.9E-01	8	3.2E-05	1.0E-07	3.2E-12
208	SS	EMFW	3	1	0	1	3.2E-05	TBVG10	1.0E-02	10	3.2E-07	1.0E-05	3.2E-12
209	SS	EMFW	3	1	0	2	8.4E-05	TBVG8	9.9E-01	8	8.3E-05	1.0E-07	8.3E-12
210	SS	EMFW	3	1	0	2	8.4E-05	TBVG10	1.0E-02	10	8.4E-07	1.0E-05	8.4E-12
211	SS	EMFW	3	1	1	0	1.4E-06	TBVG8	9.9E-01	8	1.4E-06	1.0E-07	1.4E-13
212	SS	EMFW	3	1	1	0	1.4E-06	TBVG9	1.0E-02	9	1.4E-08	2.0E-03	2.8E-11

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REC TYPE INIT  SPC  FWS  ECC  PPC   EFREQ    TRANS     HF     TRANS#  TFREQ    P(F T)   P(TWC)
213  SS  EMFW    3    3    0    0  1.8E-06   TBVG2   9.9E-01      2  1.8E-06  1.0E-07  1.8E-13
214  SS  EMFW    3    3    0    0  1.8E-06   LANL7   1.0E-02     41  1.6E-08  1.6E-03  3.2E-11
215  SS  EMFW    3    4    0    0  7.5E-06   TBVG2   9.9E-01      2  7.4E-06  1.0E-07  7.4E-13
216  SS  EMFW    3    4    0    0  7.5E-06  OVRFD6   1.0E-02     21  7.5E-08  1.4E-06  1.1E-13
217  SS  EMFW    3    4    0    0  7.5E-06  OVRFD7   3.6E-03     22  2.7E-08  1.0E-07  2.7E-15
218  SS  EMFW    3    2    0    0  4.7E-06   TBVG3   9.9E-01      3  4.7E-06  1.0E-07  4.7E-13
219  SS  EMFW    3    2    0    0  4.7E-06   TBVG4   1.0E-02      4  4.7E-08  2.0E-03  9.4E-11
220  SS  EMFW    4    1    0    0  8.6E-06   TBVG2   9.9E-01      2  8.5E-06  1.0E-07  8.5E-13
221  SS  EMFW    4    1    0    0  8.6E-06   TBVG9   1.0E-02      9  8.6E-08  2.0E-03  1.7E-10
222  SS  EMFW    5    1    0    0  6.3E-06   TBVG2   9.9E-01      2  6.2E-06  1.0E-07  6.2E-13
223  SS  EMFW    5    1    0    0  6.3E-06   TBVG9   1.0E-02      9  6.3E-08  2.0E-03  1.3E-10
224  SS  EMFW    6    1    0    0  7.0E-06   MSLB7   9.9E-01     32  6.9E-06  6.2E-04  4.3E-09
225  SS  EMFW    6    1    0    0  7.0E-06   TBVG9   1.0E-02      9  7.0E-08  2.0E-03  1.4E-10
226  RES EMFW   99   99   99   99  7.2E-05    DEF    1.0E+00     99  7.2E-05  5.4E-03  3.9E-07
227  SL  LSLB    1    0    0    0  9.6E-04   MSLB1   9.9E-01     26  9.5E-04  6.2E-04  5.9E-07
228  SL  LSLB    1    0    0    0  9.6E-04   TBVG9   1.0E-02      9  9.6E-06  2.0E-03  1.9E-08
229  SL  LSLB    1    1    0    0  3.0E-05   MSLB3   9.9E-01     28  3.0E-05  4.0E-04  1.2E-08
230  SL  LSLB    1    1    0    0  3.0E-05   TBVG9   1.0E-02      9  3.0E-07  2.0E-03  6.0E-10
231  SL  LSLB    3    0    0    0  5.5E-06   MSLB1   9.9E-01     26  5.4E-06  6.2E-04  3.4E-09
232  SL  LSLB    3    0    0    0  5.5E-06   TBVG9   1.0E-02      9  5.5E-08  2.0E-03  1.1E-10
233  RES LSLB   99   99   99   99  5.3E-06    DEF    1.0E+00     99  5.3E-06  5.4E-03  2.9E-08
234  SL  SSLB    1    0    0    0  9.0E-03   TBVG2   9.9E-01      2  8.9E-03  1.0E-07  8.9E-10
235  SL  SSLB    1    0    0    0  9.0E-03   TBVG1   1.0E-02      1  9.0E-05  4.0E-07  3.6E-11
236  SL  SSLB    1    1    0    0  2.8E-04   TBVG2   9.9E-01      2  2.8E-04  1.0E-07  2.8E-11
237  SL  SSLB    1    1    0    0  2.8E-04   TBVG1   1.0E-02      1  2.8E-06  4.0E-07  1.1E-12
238  SS  SSLB    1    4    0    0  2.4E-06   TBVG2   9.9E-01      2  2.4E-06  1.0E-07  2.4E-13
239  SS  SSLB    1    4    0    0  2.4E-06  OVRFD6   1.0E-02     21  2.4E-08  1.4E-06  3.4E-14
240  SS  SSLB    1    4    0    0  2.4E-06  OVRFD7   3.6E-03     22  8.6E-09  1.0E-07  8.6E-16
241  SL  SSLB    3    0    0    0  6.3E-04   TBVG8   9.9E-01      8  6.2E-04  1.0E-07  6.2E-11
242  SL  SSLB    3    0    0    0  6.3E-04   TBVG5   1.0E-02      5  6.3E-06  5.2E-06  3.3E-11
243  SL  SSLB    3    0    0    1  1.5E-05   TBVG8   9.9E-01      8  1.5E-05  1.0E-07  1.5E-12
244  SL  SSLB    3    0    0    1  1.5E-05  TBVG10   1.0E-02     10  1.5E-07  1.0E-05  1.5E-12
245  SL  SSLB    3    0    0    2  4.0E-05   TBVG8   9.9E-01      8  4.0E-05  1.0E-07  4.0E-12
246  SL  SSLB    3    0    0    2  4.0E-05  TBVG10   1.0E-02     10  4.0E-07  1.0E-05  4.0E-12
247  SL  SSLB    3    1    0    0  2.0E-05   TBVG8   9.9E-01      8  2.0E-05  1.0E-07  2.0E-12
248  SL  SSLB    3    1    0    0  2.0E-05   TBVG9   1.0E-02      9  2.0E-07  2.0E-03  4.0E-10
249  SL  SSLB    3    1    0    2  1.0E-06   TBVG8   9.9E-01      8  9.9E-07  1.0E-07  9.9E-14
250  SL  SSLB    3    1    0    2  1.0E-06  TBVG10   1.0E-02     10  1.0E-08  1.0E-05  1.0E-13
251  SL  SSLB    4    0    0    0  6.5E-06   TBVG1   1.0E-02      1  6.5E-08  4.0E-07  2.6E-14
252  SL  SSLB    4    0    0    0  6.5E-06   TBVG2   9.9E-01      2  6.4E-06  1.0E-07  6.4E-13
253  RES SSLB   99   99   99   99  1.4E-05    DEF    1.0E+00     99  1.4E-05  5.4E-03  7.6E-08
254 NORM LOMFW   0    0    0    0  5.5E-01   NONE    1.0E+00      0  5.5E-01  0.0E+00  0.0E+00
255 LOCA LOMFW   0    0    0    1  1.4E-02   PSBG1   1.0E+00     11  1.4E-02  1.0E-07  1.4E-09
256 LOCA LOMFW   0    0    0    2  3.6E-02   PSBG1   1.0E+00     11  3.6E-02  1.0E-07  3.6E-09
257  FW  LOMFW   0    3    0    0  2.0E-03    MIN    9.9E-01     98  2.0E-03  1.0E-07  2.0E-10
258  FW  LOMFW   0    3    0    0  2.0E-03   INEL3   1.0E-02     53  2.0E-05  1.0E-07  2.0E-12
259  FW  LOMFW   0    3    0    1  5.1E-05   PSBG5   9.9E-01     15  5.0E-05  1.0E-07  5.0E-12
260  FW  LOMFW   0    3    0    1  5.1E-05   PSBG4   1.0E-02     14  5.1E-07  1.3E-03  6.4E-10
261  FW  LOMFW   0    3    0    2  1.3E-04   PSBG5   9.9E-01     15  1.3E-04  1.0E-07  1.3E-11
262  FW  LOMFW   0    3    0    2  1.3E-04   PSBG4   1.0E-02     14  1.3E-06  1.3E-03  1.6E-09
263  FW  LOMFW   0    4    0    0  3.9E-03   NONE    9.9E-01      0  3.9E-03  0.0E+00  0.0E+00
264  FW  LOMFW   0    4    0    0  3.9E-03  OVRFD6   1.0E-02     21  3.9E-05  1.4E-06  5.5E-11
265  FW  LOMFW   0    4    0    0  3.9E-03  OVRFD7   3.6E-03     22  1.4E-05  1.0E-07  1.4E-12
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REC	TYPE	INIT	SPC	FWS	ECC	PPC	EFREQ	TRANS	HF	TRANS#	TFREQ	P(F!T)	P(TWC)
266	FW	LOMFW	0	4	0	1	9.5E-05	NONE	9.9E-01	0	9.4E-05	0.0E+00	0.0E+00
267	FW	LOMFW	0	4	0	1	9.5E-05	OVRFD6	1.0E-02	21	9.5E-07	1.4E-06	1.3E-12
268	FW	LOMFW	0	4	0	1	9.5E-05	OVRFD7	3.6E-03	22	3.4E-07	1.0E-07	3.4E-14
269	FW	LOMFW	0	4	0	2	2.5E-04	NONE	9.9E-01	0	2.5E-04	0.0E+00	0.0E+00
270	FW	LOMFW	0	4	0	2	2.5E-04	OVRFD6	1.0E-02	21	2.5E-06	1.4E-06	3.5E-12
271	FW	LOMFW	0	4	0	2	2.5E-04	OVRFD7	3.6E-03	22	9.0E-07	1.0E-07	9.0E-14
272	SL	LOMFW	1	0	0	0	8.4E-02	TBVG2	9.9E-01	2	8.3E-02	1.0E-07	8.3E-09
273	SL	LOMFW	1	0	0	0	8.4E-02	TBVG1	1.0E-02	1	8.4E-04	4.0E-07	3.4E-10
274	SL	LOMFW	1	0	0	1	2.1E-03	TBVG2	9.9E-01	2	2.1E-03	1.0E-07	2.1E-10
275	SL	LOMFW	1	0	0	1	2.1E-03	TBVG6	1.0E-02	6	2.1E-05	1.7E-06	3.6E-11
276	SL	LOMFW	1	0	0	2	5.5E-03	TBVG2	9.9E-01	2	5.4E-03	1.0E-07	5.4E-10
277	SL	LOMFW	1	0	0	2	5.5E-03	TBVG6	1.0E-02	6	5.5E-05	1.7E-06	9.4E-11
278	SL	LOMFW	1	0	1	0	1.1E-04	TBVG2	9.9E-01	2	1.1E-04	1.0E-07	1.1E-11
279	SL	LOMFW	1	0	1	0	1.1E-04	TBVG1	1.0E-02	1	1.1E-06	4.0E-07	4.4E-13
280	SL	LOMFW	1	0	1	1	2.7E-06	TBVG2	9.9E-01	2	2.7E-06	1.0E-07	2.7E-13
281	SL	LOMFW	1	0	1	1	2.7E-06	TBVG6	1.0E-02	6	2.7E-08	1.7E-06	4.6E-14
282	SL	LOMFW	1	0	1	2	6.2E-06	TBVG2	9.9E-01	2	6.1E-06	1.0E-07	6.1E-13
283	SL	LOMFW	1	0	1	2	6.2E-06	TBVG6	1.0E-02	6	6.2E-08	1.7E-06	1.1E-13
284	SS	LOMFW	1	3	0	0	3.1E-04	TBVG2	9.9E-01	2	3.1E-04	1.0E-07	3.1E-11
285	SS	LOMFW	1	3	0	0	3.1E-04	LANL7	1.0E-02	41	3.1E-06	1.8E-03	5.6E-09
286	SS	LOMFW	1	3	0	1	7.0E-06	PSBG5	9.9E-01	15	6.9E-06	1.0E-07	6.9E-13
287	SS	LOMFW	1	3	0	1	7.0E-06	TBVG6	1.0E-02	6	7.0E-08	1.7E-06	1.2E-13
288	SS	LOMFW	1	3	0	2	1.6E-05	PSBG5	9.9E-01	15	1.6E-05	1.0E-07	1.6E-12
289	SS	LOMFW	1	3	0	2	1.6E-05	TBVG6	1.0E-02	6	1.6E-07	1.7E-06	2.7E-13
290	SS	LOMFW	1	4	0	0	5.9E-04	TBVG2	9.9E-01	2	5.8E-04	1.0E-07	5.8E-11
291	SS	LOMFW	1	4	0	0	5.9E-04	OVRFD6	1.0E-02	21	5.9E-06	1.4E-06	8.3E-12
292	SS	LOMFW	1	4	0	0	5.9E-04	OVRFD7	3.6E-03	22	2.1E-06	1.0E-07	2.1E-13
293	SS	LOMFW	1	4	0	1	1.5E-05	PSBG5	9.9E-01	15	1.5E-05	1.0E-07	1.5E-12
294	SS	LOMFW	1	4	0	1	1.5E-05	OVRFD6	1.0E-02	21	1.5E-07	1.4E-06	2.1E-13
295	SS	LOMFW	1	4	0	1	1.5E-05	OVRFD7	3.6E-03	22	5.4E-08	1.0E-07	5.4E-15
296	SS	LOMFW	1	4	0	2	3.7E-05	PSBG5	9.9E-01	15	3.7E-05	1.0E-07	3.7E-12
297	SS	LOMFW	1	4	0	2	3.7E-05	OVRFD6	1.0E-02	21	3.7E-07	1.4E-06	5.2E-13
298	SS	LOMFW	1	4	0	2	3.7E-05	OVRFD7	3.6E-03	22	1.3E-07	1.0E-07	1.3E-14
299	SL	LOMFW	2	0	0	0	9.8E-04	TBVG2	9.9E-01	0	9.7E-04	0.0E+00	0.0E+00
300	SL	LOMFW	2	0	0	0	9.8E-04	TBVG1	1.0E-02	0	9.8E-06	0.0E+00	0.0E+00
301	SL	LOMFW	2	0	0	1	2.4E-05	PSBG5	9.9E-01	15	2.4E-05	1.0E-07	2.4E-12
302	SL	LOMFW	2	0	0	1	2.4E-05	TBVG6	1.0E-02	6	2.4E-07	1.7E-06	4.1E-13
303	SL	LOMFW	2	0	0	2	6.3E-05	PSBG5	9.9E-01	15	6.2E-05	1.0E-07	6.2E-12
304	SL	LOMFW	2	0	0	2	6.3E-05	TBVG6	1.0E-02	6	6.3E-07	1.7E-06	1.1E-12
305	SS	LOMFW	2	3	0	0	3.0E-06	TBVG2	9.9E-01	2	3.0E-06	1.0E-07	3.0E-13
306	SS	LOMFW	2	3	0	0	3.0E-06	LANL7	1.0E-02	41	3.0E-08	1.8E-03	5.4E-11
307	SS	LOMFW	2	4	0	0	5.0E-06	TBVG2	9.9E-01	2	5.0E-06	1.0E-07	5.0E-13
308	SS	LOMFW	2	4	0	0	5.0E-06	OVRFD6	1.0E-02	21	5.0E-08	1.4E-06	7.0E-14
309	SS	LOMFW	2	4	0	0	5.0E-06	OVRFD7	3.6E-03	22	1.8E-06	1.0E-07	1.8E-15
310	SL	LOMFW	3	0	0	0	9.2E-03	TBVG8	9.9E-01	8	9.1E-03	1.0E-07	9.1E-10
311	SL	LOMFW	3	0	0	0	9.2E-03	TBVG5	1.0E-02	5	9.2E-05	5.2E-06	4.8E-10
312	SL	LOMFW	3	0	0	1	2.3E-04	TBVG8	9.9E-01	8	2.3E-04	1.0E-07	2.3E-11
313	SL	LOMFW	3	0	0	1	2.3E-04	TBVG10	1.0E-02	10	2.3E-06	1.0E-05	2.3E-11
314	SL	LOMFW	3	0	0	2	6.0E-04	TBVG8	9.9E-01	8	5.9E-04	1.0E-07	5.9E-11
315	SL	LOMFW	3	0	0	2	6.0E-04	TBVG10	1.0E-02	10	6.0E-06	1.0E-05	6.0E-11
316	SL	LOMFW	3	0	1	0	1.2E-05	TBVG8	9.9E-01	8	1.2E-05	1.0E-07	1.2E-12
317	SL	LOMFW	3	0	1	0	1.2E-05	TBVG5	1.0E-02	5	1.2E-07	5.2E-06	6.2E-13
318	SS	LOMFW	3	3	0	0	3.4E-05	TBVG2	9.9E-01	2	3.4E-05	1.0E-07	3.4E-12

REC	TYPE	INIT	SPC	FWS	ECC	PPC	EFREQ	TRANS	HF	TRANS#	TFREQ	P(F|T)	P(TWC)
319	SS	LOMFW	3	3	0	0	3.4E-05	LANL7	1.0E-02	41	3.4E-07	1.8E-03	6.1E-10
320	SS	LOMFW	3	3	0	2	1.8E-06	PSBG5	9.9E-01	15	1.8E-06	1.0E-07	1.8E-13
321	SS	LOMFW	3	3	0	2	1.8E-06	TBVG6	1.0E-02	6	1.8E-08	1.7E-06	3.1E-14
322	SS	LOMFW	3	4	0	0	6.5E-05	TBVG2	9.9E-01	2	6.4E-05	1.0E-07	6.4E-12
323	SS	LOMFW	3	4	0	0	6.5E-05	OVRFD6	1.0E-02	21	6.5E-07	1.4E-06	9.1E-13
324	SS	LOMFW	3	4	0	0	6.5E-05	OVRFD7	3.6E-03	22	2.3E-07	1.0E-07	2.3E-14
325	SS	LOMFW	3	4	0	1	1.3E-06	PSBG5	9.9E-01	15	1.3E-06	1.0E-07	1.3E-13
326	SS	LOMFW	3	4	0	1	1.3E-06	OVRFD6	1.0E-02	21	1.3E-08	1.4E-06	1.8E-14
327	SS	LOMFW	3	4	0	1	1.3E-06	OVRFD7	3.6E-03	22	4.7E-09	1.0E-07	4.7E-16
328	SS	LOMFW	3	4	0	2	3.0E-06	PSBG5	9.9E-01	15	3.0E-06	1.0E-07	3.0E-13
329	SS	LOMFW	3	4	0	2	3.0E-06	OVRFD6	1.0E-02	21	3.0E-08	1.4E-06	4.2E-14
330	SS	LOMFW	3	4	0	2	3.0E-06	OVRFD7	3.6E-03	22	1.1E-08	1.0E-07	1.1E-15
331	SL	LOMFW	4	0	0	0	8.3E-05	TBVG2	9.9E-01	2	8.2E-05	1.0E-07	8.2E-12
332	SL	LOMFW	4	0	0	0	8.3E-05	TBVG1	1.0E-02	1	8.3E-07	4.0E-07	3.3E-13
333	SL	LOMFW	4	0	0	2	4.3E-06	TBVG2	9.9E-01	2	4.3E-06	1.0E-07	4.3E-13
334	SL	LOMFW	4	0	0	2	4.3E-06	TBVG10	1.0E-02	10	4.3E-08	1.0E-05	4.3E-13
335	SL	LOMFW	5	0	0	0	5.4E-05	TBVG2	9.9E-01	2	5.3E-05	1.0E-07	5.3E-12
336	SL	LOMFW	5	0	0	0	5.4E-05	TBVG5	1.0E-02	5	5.4E-07	5.2E-06	2.8E-12
337	SL	LOMFW	5	0	0	1	1.3E-06	TBVG2	9.9E-01	2	1.3E-06	1.0E-07	1.3E-13
338	SL	LOMFW	5	0	0	1	1.3E-06	TBVG10	1.0E-02	10	1.3E-08	1.0E-05	1.3E-13
339	SL	LOMFW	5	0	0	2	3.1E-06	TBVG2	9.9E-01	2	3.1E-06	1.0E-07	3.1E-13
340	SL	LOMFW	5	0	0	2	3.1E-06	TBVG10	1.0E-02	10	3.1E-08	1.0E-05	3.1E-13
341	SL	LOMFW	6	0	0	0	6.1E-05	MSLB1	9.9E-01	26	6.0E-05	6.2E-04	3.7E-08
342		LOMFW	6	0	0	0	6.1E-05	MSLB1	9.9E-01	26	6.0E-05	6.2E-04	3.7E-08
343		LOMFW	6	0	0	0	6.1E-05	TBVG9	1.0E-02	9	6.1E-07	2.0E-03	1.2E-09
344	SL	LOMFW	6	0	0	1	1.5E-06	MSLB7	9.9E-01	32	1.5E-06	6.2E-04	9.2E-10
345	SL	LOMFW	6	0	0	1	1.5E-06	TBVG9	1.0E-02	9	1.5E-08	2.0E-03	3.0E-11
346	SL	LOMFW	6	0	0	2	3.5E-06	MSLB7	9.9E-01	32	3.5E-06	6.2E-04	2.1E-09
347	SL	LOMFW	6	0	0	2	3.5E-06	TBVG9	1.0E-02	9	3.5E-08	2.0E-03	7.0E-11
348	RES	LOMFW	99	99	99	99	5.9E-05	DEF	1.0E+00	99	5.9E-05	5.4E-03	3.2E-07
349	LOCA	SBL1	0	0	0	0	7.6E-02	PSBG1	1.0E+00	11	7.6E-02	1.0E-07	7.6E-09
350	LOCA	SBL1	0	0	0	1	1.9E-03	PSBG1	1.0E+00	11	1.9E-03	1.0E-07	1.9E-10
351	LOCA	SBL1	0	0	0	2	5.0E-03	PSBG1	1.0E+00	11	5.0E-03	1.0E-07	5.0E-10
352	LOCA	SBL1	0	0	1	0	9.9E-05	PSBG1	1.0E+00	11	9.9E-05	1.0E-07	9.9E-12
353	LOCA	SBL1	0	0	1	1	2.4E-06	PSBG1	1.0E+00	11	2.4E-06	1.0E-07	2.4E-13
354	LOCA	SBL1	0	0	1	2	5.5E-06	PSBG1	1.0E+00	11	5.5E-06	1.0E-07	5.5E-13
355	LOCA	SBL1	0	1	0	0	2.4E-03	PSBG1	1.0E+00	11	2.4E-03	1.0E-07	2.4E-10
356	LOCA	SBL1	0	1	0	1	5.9E-05	PSBG1	1.0E+00	11	5.9E-05	1.0E-07	5.9E-12
357	LOCA	SBL1	0	1	0	2	1.6E-04	PSBG5	1.0E+00	15	1.6E-04	1.0E-07	1.6E-11
358	LOCA	SBL1	0	1	1	0	2.5E-06	PSBG5	1.0E+00	15	2.5E-06	1.0E-07	2.5E-13
359	LOCA	SBL1	0	3	0	0	6.6E-06	PSBG5	9.9E-01	15	6.5E-06	1.0E-07	6.5E-13
360	LOCA	SBL1	0	3	0	0	6.6E-06	PSBG4	1.0E-02	14	6.6E-08	1.3E-03	8.3E-11
361	LOCA	SBL1	0	4	0	0	3.1E-05	NONE	9.9E-01	0	3.1E-05	0.0E+00	0.0E+00
362	LOCA	SBL1	0	4	0	0	3.1E-05	OVRFD6	1.0E-02	21	3.1E-07	1.4E-06	4.3E-13
363	LOCA	SBL1	0	4	0	0	3.1E-05	OVRFD7	3.6E-03	22	1.1E-07	1.0E-07	1.1E-14
364	LOCA	SBL1	0	2	0	0	8.7E-06	TBVG2	9.9E-01	2	8.6E-06	1.0E-07	8.6E-13
365	LOCA	SBL1	0	2	0	0	8.7E-06	TBVG6	1.0E-02	6	8.7E-08	1.7E-06	1.5E-13
366	LOCA	SBL1	1	0	0	0	1.2E-02	TBVG2	9.9E-01	2	1.2E-02	1.0E-07	1.2E-09
367	LOCA	SBL1	1	0	0	0	1.2E-02	TBVG6	1.0E-02	6	1.2E-04	1.7E-06	2.0E-10
368	LOCA	SBL1	1	0	0	1	2.9E-04	PSBG5	9.9E-01	15	2.9E-04	1.0E-07	2.9E-11
369	LOCA	SBL1	1	0	0	1	2.9E-04	TBVG6	1.0E-02	6	2.9E-06	1.7E-06	4.9E-12
370	LOCA	SBL1	1	0	0	2	7.5E-04	PSBG5	9.9E-01	15	7.4E-04	1.0E-07	7.4E-11
371	LOCA	SBL1	1	0	0	2	7.5E-04	TBVG6	1.0E-02	6	7.5E-06	1.7E-06	1.3E-11

BEST COPY AVAIL

REC	TYPE	INIT	SPC	FWS	ECC	PFC	EFREQ	TRANS.	MF	TRANS#	TFREQ	P(F!T)	P(TWC)
372	LOCA	SBL1	1	0	1	0	1.3E-05	TBVG2	9.9E-01	2	1.3E-05	1.0E-07	1.3E-12
373	LOCA	SBL1	1	0	1	0	1.3E-05	TBVG6	1.0E-02	6	1.3E-07	1.7E-06	2.2E-13
374	LOCA	SBL1	1	3	0	0	1.7E-05	PSBG5	9.9E-01	15	1.7E-05	1.0E-07	1.7E-12
375	LOCA	SBL1	1	3	0	0	1.7E-05	TBVG6	1.0E-02	6	1.7E-07	1.7E-06	2.9E-13
376	LOCA	SBL1	1	4	0	0	7.7E-05	PSBG5	9.9E-01	15	7.6E-05	1.0E-07	7.6E-12
377	LOCA	SBL1	1	4	0	0	7.7E-05	OVRFD6	1.0E-02	21	7.7E-07	1.4E-06	1.1E-12
378	LOCA	SBL1	1	4	0	2	3.6E-06	PSBG5	9.9E-01	15	3.6E-06	1.0E-07	3.6E-13
379	LOCA	SBL1	1	4	0	2	3.6E-06	OVRFD6	1.0E-02	21	3.6E-08	1.4E-06	5.0E-14
380	LOCA	SBL1	1	4	0	2	3.6E-06	OVRFD7	3.6E-03	22	1.3E-08	1.0E-07	1.3E-15
381	LOCA	SBL1	1	1	0	0	3.7E-04	PSBG5	9.9E-01	15	3.7E-04	1.0E-07	3.7E-11
382	LOCA	SBL1	1	1	0	0	3.7E-04	TBVG6	1.0E-02	6	3.7E-06	1.7E-06	6.3E-12
383	LOCA	SBL1	1	1	0	1	8.2E-06	PSBG5	9.9E-01	15	8.1E-06	1.0E-07	8.1E-13
384	LOCA	SBL1	1	1	0	1	8.2E-06	TBVG6	1.0E-02	6	8.2E-08	1.7E-06	1.4E-13
385	LOCA	SBL1	1	1	0	2	2.1E-05	PSBG5	9.9E-01	15	2.1E-05	1.0E-07	2.1E-12
386	LOCA	SBL1	1	1	0	2	2.1E-05	TBVG6	1.0E-02	6	2.1E-07	1.7E-06	3.6E-13
387	LOCA	SBL1	2	0	0	0	1.4E-04	PSBG5	9.9E-01	15	1.4E-04	1.0E-07	1.4E-11
388	LOCA	SBL1	2	0	0	0	1.4E-04	TBVG6	1.0E-02	6	1.4E-06	1.7E-06	2.4E-12
389	LOCA	SBL1	2	0	0	2	7.6E-06	PSBG5	9.9E-01	15	7.5E-06	1.0E-07	7.5E-13
390	LOCA	SBL1	2	0	0	2	7.6E-06	TBVG6	1.0E-02	6	7.6E-08	1.7E-06	1.3E-13
391	LOCA	SBL1	2	1	0	0	3.5E-06	TBVG3	9.9E-01	3	3.5E-06	1.0E-07	3.5E-13
392	LOCA	SBL1	2	1	0	0	3.5E-06	TBVG4	1.0E-02	4	3.5E-08	2.0E-03	7.0E-11
393	LOCA	SBL1	3	0	0	0	1.3E-03	TBVG8	9.9E-01	8	1.3E-03	1.0E-07	1.3E-10
394	LOCA	SBL1	3	0	0	0	1.3E-03	TBVG10	1.0E-02	10	1.3E-05	1.0E-05	1.3E-10
395	LOCA	SBL1	3	0	0	1	3.0E-05	TBVG8	9.9E-01	8	3.0E-05	1.0E-07	3.0E-12
396	LOCA	SBL1	3	0	0	1	3.0E-05	TBVG10	1.0E-02	10	3.0E-07	1.0E-05	3.0E-12
397	LOCA	SBL1	3	0	0	2	8.3E-05	TBVG8	9.9E-01	8	8.2E-05	1.0E-07	8.2E-12
398	LOCA	SBL1	3	0	0	2	8.3E-05	TBVG10	1.0E-02	10	8.3E-07	1.0E-05	8.3E-12
399	LOCA	SBL1	3	0	1	0	1.4E-06	TBVG8	9.9E-01	8	1.4E-06	1.0E-07	1.4E-13
400	LOCA	SBL1	3	0	1	0	1.4E-06	TBVG10	1.0E-02	10	1.4E-08	1.0E-05	1.4E-13
401	LOCA	SBL1	3	3	0	0	1.9E-06	TBVG2	9.9E-01	2	1.9E-06	1.0E-07	1.9E-13
402	LOCA	SBL1	3	3	0	0	1.9E-06	LANL7	1.0E-02	41	1.9E-08	1.8E-03	3.4E-11
403	LOCA	SBL1	3	4	0	0	7.8E-06	TBVG2	9.9E-01	2	7.7E-06	1.0E-07	7.7E-13
404	LOCA	SBL1	3	4	0	0	7.8E-06	OVRFD6	1.0E-02	21	7.8E-08	1.4E-06	1.1E-13
405	LOCA	SBL1	3	4	0	0	7.8E-06	OVRFD7	3.6E-03	22	2.8E-08	1.0E-07	2.8E-15
406	LOCA	SBL1	3	1	0	0	4.0E-05	TBVG8	9.9E-01	8	4.0E-05	1.0E-07	4.0E-12
407	LOCA	SBL1	3	1	0	0	4.0E-05	TBVG10	1.0E-02	10	4.0E-07	1.0E-05	4.0E-12
408	LOCA	SBL1	3	1	0	2	2.1E-06	TBVG8	9.9E-01	8	2.1E-06	1.0E-07	2.1E-13
409	LOCA	SBL1	3	1	0	2	2.1E-06	TBVG10	1.0E-02	10	2.1E-08	1.0E-05	2.1E-13
410	LOCA	SBL1	4	0	0	0	9.1E-06	TBVG2	9.9E-01	2	9.0E-06	1.0E-07	9.0E-13
411	LOCA	SBL1	4	0	0	0	9.1E-06	TBVG10	1.0E-02	10	9.1E-08	1.0E-05	9.1E-13
412	LOCA	SBL1	5	0	0	0	6.6E-06	TBVG2	9.9E-01	2	6.5E-06	1.0E-07	6.5E-13
413	LOCA	SBL1	5	0	0	0	6.6E-06	TBVG10	1.0E-02	10	6.6E-08	1.0E-05	6.6E-13
414	LOCA	SBL1	6	0	0	0	7.3E-06	MSLB7	9.9E-01	32	7.2E-06	6.2E-04	4.5E-09
415	LOCA	SBL1	6	0	0	0	7.3E-06	TBVG9	1.0E-02	9	7.3E-08	2.0E-03	1.5E-10
416	RES	SBL1	99	99	99	99	7.8E-05	DEF	1.0E+00	99	7.8E-05	5.4E-03	4.2E-07
417	LOCA	SBL2	0	0	0	0	8.2E-03	PSBG1	1.0E+00	11	8.2E-03	1.0E-07	8.2E-10
418	LOCA	SBL2	0	0	1	0	1.0E-05	PSBG1	1.0E+00	11	1.0E-05	1.0E-07	1.0E-12
419	LOCA	SBL2	0	1	0	0	2.6E-04	PSBG5	9.9E-01	15	2.6E-04	1.0E-07	2.6E-11
420	LOCA	SBL2	0	1	0	0	2.6E-04	PSBG4	1.0E-02	14	2.6E-06	1.3E-03	3.3E-09
421	LOCA	SBL2	0	4	0	0	2.7E-06	NONE	9.9E-01	0	2.7E-06	0.0E+00	0.0E+00
422	LOCA	SBL2	0	4	0	0	2.7E-06	OVRFD6	1.0E-02	21	2.7E-08	1.4E-06	3.8E-14
423	LOCA	SBL2	0	4	0	0	2.7E-06	OVRFD7	3.6E-03	22	9.7E-09	1.0E-07	9.7E-15
424	LOCA	SBL2	0	2	0	0	1.0E-06	DEF	1.0E+00	99	1.0E-06	5.4E-03	5.4E-09

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REC TYPE INIT  SPC  FWS  ECC  PFC   EFREQ     TRANS     HF     TRANS#  TFREQ    P(F:T)    P(T#C)
425 LOCA SBL2    1    0    0    0  1.2E-03    TBVG2   9.9E-01      2  1.2E-03  1.0E-07  1.2E-10
426 LOCA SBL2    1    0    0    0  1.2E-03    TBVG6   1.0E-02      6  1.2E-05  1.7E-06  2.0E-11
427 LOCA SBL2    1    3    0    0  2.1E-06    PSBG5   9.9E-01     15  2.1E-06  1.0E-07  2.1E-13
428 LOCA SBL2    1    3    0    0  2.1E-06    TBVG6   1.0E-02      6  2.1E-08  1.7E-06  3.6E-14
429 LOCA SBL2    1    4    0    0  6.9E-06    PSBG5   9.9E-01     15  6.8E-06  1.0E-07  6.8E-13
430 LOCA SBL2    1    4    0    0  6.9E-06   OVRFD6   1.0E-02     21  6.9E-08  1.4E-06  9.7E-14
431 LOCA SBL2    1    4    0    0  6.9E-06   OVRFD7   3.6E-03     22  2.5E-08  1.0E-07  2.5E-15
432 LOCA SBL2    1    1    0    0  4.0E-05    PSBG5   9.9E-01     15  4.0E-05  1.0E-07  4.0E-12
433 LOCA SBL2    1    1    0    0  4.0E-05    TBVG6   1.0E-02      6  4.0E-07  1.7E-06  6.8E-13
434 LOCA SBL2    2    0    0    0  1.4E-05    PSBG5   9.9E-01     15  1.4E-05  1.0E-07  1.4E-12
435 LOCA SBL2    2    0    0    0  1.4E-05    TBVG6   1.0E-02      6  1.4E-07  1.7E-06  2.4E-13
436 LOCA SBL2    3    0    0    0  1.4E-04    TBVG8   9.9E-01      8  1.4E-04  1.0E-07  1.4E-11
437 LOCA SBL2    3    0    0    0  1.4E-04   TBVG10   1.0E-02     10  1.4E-06  1.0E-05  1.4E-11
438 LOCA SBL2    3    1    0    0  4.0E-06    TBVG8   9.9E-01      8  4.0E-06  1.0E-07  4.0E-13
439 LOCA SBL2    3    1    0    0  4.0E-06   TBVG10   1.0E-02     10  4.0E-08  1.0E-05  4.0E-13
440  RES SBL2   99   99   99   99  1.8E-05     DEF    1.0E+00     99  1.8E-05  5.4E-03  9.9E-08
441 SGTR SGTR    0    0    0    0  8.2E-03    SGTR    1.0E+00     23  8.2E-03  1.0E-07  8.2E-10
442 SGTR SGTR    0    0    1    0  1.0E-05    SGTR    1.0E+00     23  1.0E-05  1.0E-07  1.0E-12
443 SGTR SGTR    0    1    0    0  2.6E-04    PSBG5   9.9E-01     15  2.6E-04  1.0E-07  2.6E-11
444 SGTR SGTR    0    1    0    0  2.6E-04    PSBG4   1.0E-02     14  2.6E-06  1.3E-03  3.3E-09
445 SGTR SGTR    0    4    0    0  2.7E-06    NONE    9.7E-01      0  2.7E-06  0.0E+00  0.0E+00
446 SGTR SGTR    0    4    0    0  2.7E-06   OVRFD6   1.0E-02     21  2.7E-08  1.4E-06  3.8E-14
447 SGTR SGTR    0    4    0    0  2.7E-06   OVRFD7   3.6E-03     22  9.7E-09  1.0E-07  9.7E-16
448 SGTR SGTR    0    2    0    0  1.0E-06    TBVG2   9.9E-01      2  9.9E-07  1.0E-07  9.9E-14
449 SGTR SGTR    0    2    0    0  1.0E-06    TBVG6   1.0E-02      6  1.0E-08  1.7E-06  1.7E-14
450 SGTR SGTR    1    0    0    0  1.2E-03    TBVG2   9.9E-01      2  1.2E-03  1.0E-07  1.2E-10
451 SGTR SGTR    1    0    0    0  1.2E-03    TBVG6   1.0E-02      6  1.2E-05  1.7E-06  2.0E-11
452 SGTR SGTR    1    3    0    0  2.1E-06    PSBG5   9.9E-01     15  2.1E-06  1.0E-07  2.1E-13
453 SGTR SGTR    1    3    0    0  2.1E-06    TBVG6   1.0E-02      6  2.1E-08  1.7E-06  3.6E-14
454 SGTR SGTR    1    4    0    0  6.9E-06    PSBG5   9.9E-01     15  6.8E-06  1.0E-07  6.8E-13
455 SGTR SGTR    1    4    0    0  6.9E-06   OVRFD6   1.0E-02     21  6.9E-08  1.4E-06  9.7E-14
456 SGTR SGTR    1    4    0    0  6.9E-06   OVRFD7   3.6E-03     22  2.5E-08  1.0E-07  2.5E-15
457 SGTR SGTR    1    1    0    0  4.0E-05    PSBG5   9.9E-01     15  4.0E-05  1.0E-07  4.0E-12
458 SGTR SGTR    1    1    0    0  4.0E-05    TBVG6   1.0E-02      6  4.0E-07  1.7E-06  6.8E-13
459 SGTR SGTR    2    0    0    0  1.4E-05    PSBG5   9.9E-01     15  1.4E-05  1.0E-07  1.4E-12
460 SGTR SGTR    2    0    0    0  1.4E-05    TBVG6   1.0E-02      6  1.4E-07  1.7E-06  2.4E-13
461 SGTR SGTR    3    0    0    0  1.4E-04    TBVG8   9.9E-01      8  1.4E-04  1.0E-07  1.4E-11
462 SGTR SGTR    3    0    0    0  1.4E-04   TBVG10   1.0E-02     10  1.4E-06  1.0E-05  1.4E-11
463 SGTR SGTR    3    1    0    0  4.0E-06    TBVG8   9.9E-01      8  4.0E-06  1.0E-07  4.0E-13
464 SGTR SGTR    3    1    0    0  4.0E-06   TBVG10   1.0E-02     10  4.0E-08  1.0E-05  4.0E-13
465  RES SGTR   99   99   99   99  1.8E-05     DEF    1.0E+00     99  1.8E-05  5.4E-03  9.9E-08
466 NORM  SI     0    0    0    0  7.6E-03    NONE    1.0E+00      0  7.6E-03  0.0E+00  0.0E+00
467 LOCA  SI     0    0    0    1  1.9E-04    PSBG1   1.0E+00     11  1.9E-04  1.0E-07  1.9E-11
468 LOCA  SI     0    0    0    2  5.0E-04    PSBG1   1.0E+00     11  5.0E-04  1.0E-07  5.0E-11
469  FW   SI     0    1    0    0  2.4E-04   OVRFD3   1.0E+00     18  2.4E-04  1.0E-07  2.4E-11
470 LOCA  SI     0    1    0    1  5.4E-06    TBVG2   9.9E-01      2  5.3E-06  1.0E-07  5.3E-13
471 LOCA  SI     0    1    0    1  5.4E-06    TBVG6   1.0E-02      6  5.4E-08  1.7E-06  9.2E-14
472 LOCA  SI     0    1    0    2  1.5E-05    TBVG2   9.9E-01      2  1.5E-05  1.0E-07  1.5E-12
473 LOCA  SI     0    1    0    2  1.5E-05    TBVG6   1.0E-02      6  1.5E-07  1.7E-06  2.6E-13
474  FW   SI     0    4    0    0  2.2E-06    NONE    9.9E-01      0  2.2E-06  0.0E+00  0.0E+00
475  FW   SI     0    4    0    0  2.2E-06   OVRFD6   1.0E-02     21  2.2E-08  1.4E-06  3.1E-14
476  FW   SI     0    4    0    0  2.2E-06   OVRFD7   3.6E-03     22  7.9E-09  1.0E-07  7.9E-16
477  SL   SI     1    0    0    0  1.2E-03    TBVG2   9.9E-01      2  1.2E-03  1.0E-07  1.2E-10
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REC TYPE INIT  SFC  FWS  ECC  PPC   EFREQ     TRANS      HF     TRANS#  TFREQ    P(F|T)   P(TWC)
478  SL   SI     1    0    0    0  1.2E-03    TBVG1   1.0E-02      1  1.2E-05  4.0E-07  4.8E-12
479  SL   SI     1    0    0    1  2.8E-05    TBVG2   9.9E-01      2  2.8E-05  1.0E-07  2.8E-12
480  SL   SI     1    0    0    1  2.8E-05    TBVG6   1.0E-02      6  2.8E-07  1.7E-06  4.8E-13
481  SL   SI     1    0    0    2  7.2E-05    TBVG2   9.9E-01      2  7.1E-05  1.0E-07  7.1E-12
482  SL   SI     1    0    0    2  7.2E-05    TBVG6   1.0E-02      6  7.2E-07  1.7E-06  1.2E-12
483  SS   SI     1    4    0    0  5.7E-06    TBVG2   9.9E-01      2  5.6E-06  1.0E-07  5.6E-13
484  SS   SI     1    4    0    0  5.7E-06   OVRFD6   1.0E-02     21  5.7E-08  1.4E-06  8.0E-14
485  SS   SI     1    4    0    0  5.7E-06   OVRFD7   3.6E-03     22  2.1E-08  1.0E-07  2.1E-15
486  SS   SI     1    1    0    0  3.6E-05    TBVG2   9.9E-01      2  3.6E-05  1.0E-07  3.6E-12
487  SS   SI     1    1    0    0  3.6E-05    TBVG1   1.0E-02      1  3.6E-07  4.0E-07  1.4E-13
488  SL   SI     2    0    0    0  1.2E-05    TBVG2   9.9E-01      2  1.2E-05  1.0E-07  1.2E-12
489  SL   SI     2    0    0    0  1.2E-05    TBVG1   1.0E-02      1  1.2E-07  4.0E-07  4.8E-14
490  SL   SI     3    0    0    0  1.3E-04    TBVG8   9.9E-01      8  1.3E-04  1.0E-07  1.3E-11
491  SL   SI     3    0    0    0  1.3E-04    TBVG5   1.0E-02      5  1.3E-06  5.2E-06  6.6E-12
492  SL   SI     3    0    0    1  3.1E-06    TBVG8   9.9E-01      8  3.1E-06  1.0E-07  3.1E-13
493  SL   SI     3    0    0    1  3.1E-06   TBVG10   1.0E-02     10  3.1E-08  1.0E-05  3.1E-13
494  SL   SI     3    0    0    2  7.1E-06    TBVG8   9.9E-01      8  7.0E-06  1.0E-07  7.0E-13
495  SL   SI     3    0    0    2  7.1E-06   TBVG10   1.0E-02     10  7.1E-08  1.0E-05  7.1E-13
496  SS   SI     3    1    0    0  3.3E-06    TBVG8   9.9E-01      8  3.3E-06  1.0E-07  3.3E-13
497  SS   SI     3    1    0    0  3.3E-06    TBVG9   1.0E-02      9  3.3E-08  2.0E-03  6.6E-11
498  RES  SI    99   99   99   99  3.9E-05     DEF    1.0E+00     99  3.9E-05  5.4E-03  2.1E-07
```

APPENDIX B

EVENT TREE QUANTIFICATION

B.1 Introduction

This appendix presents the initial probability information used in the Oconee PTS study. While not all of the information in this appendix was used in the study, it is included for completeness.

To determine bounding values for sequence frequencies, branch probabilities were estimated for each branch on the event trees. It must be emphasized that imposed constraints permitted only simplified approaches to frequency and probability estimation, including, in several cases, the use of generic data. As with all event trees, the probability of being on a particular branch is conditional on the prior branches in the sequence. In most cases, however, branches were sufficiently independent to permit the use of common probability values in the sequence quantifications.

Probability values were estimated, when possible, from operational information which was as close to Oconee-specific as possible. If Oconee-specific information was available, then it was used. If this was not the case, B&W-specific and finally PWR-specified operational information was employed. Additional information was obtained from screening values from "IEEE Guide to the Collection and Presentation of Electrical, Electronic, and Sensing Component Reliability Data for Nuclear-Power Generating Station," IEEE Std 500-1977, sponsored by the Nuclear Power Engineering Committee of the IEEE Power Engineering Society, as well as other sources.

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Section B.2 provides development information and final probability and frequency values for system-related failures included in the event trees. Applicable sources of failure information are identified. Section B.3 describes the human reliability numbers employed on the trees. Probability values strongly impacted by thermal-hydraulic considerations are discussed in Section B.4.

B.2 System-Related Probability Values

The bases for the system-related probability values used on the event tree branches are provided in Table B.1.

B.3 Human Error-Related Probability Values

Branch probability values dominated by human performance were developed based on the "Handbook of Human Reliability Analysis" (NUREG/CR-1278), the "Generic Data Base for Data and Models Chapter of the NREP Guide" (EGG-EA-5887, June 1982), past B&W plant operational experience with steam generator overfill ("AEOD Observations and Recommendations Concerning the Problem of Steam Generator Overfill and Combined Primary and Secondary Side Blowdown," December 17, 1980), as well as requirements for operator action as described in applicable Oconee procedures.

In general, branch probabilities utilized on event tree branches dominated by human error were consistent with NUREG/CR-1278. Human error values used were:

a. For well defined tasks under little pressure that required specific action, 5×10^{-3}/D was assigned. This value was employed, for

Table B.1 System-Related Event Tree Frequencies And Branch Probabilities

Function	Discussion	Value Used
Initiators		
Reactor Trip	During 1978 and 1979, Oconee 1, 2, and 3 experienced 36 manual or automatic scrams during forced shutdowns (see NUREG 0618 and NUREG/CR-1496). This results in a non-specific reactor trip estimate of 6/year. Approximately 61% were the result of electrical or steam & power conversion system problems which, in some cases, may have impacted subsequent plant response.	6/yr
Small Break LOCA	The NREP screening value (EGG-EA-5887) for a small break LOCA, 1 x 10^{-2}, was used in this study. This value is consistent with the NUREG/CR-2497 value of 8.3 x 10^{-3}/yr for PWRs. It should be noted that this value may be somewhat high, since transient induced LOCAs are considered on the nonspecific reactor trip event tree.	10^{-2}/yr
Steam Line Break	A value equal to the small break LOCA frequency (10^{-2}/yr) was assumed. This value is applicable to a small or moderate sized steam line break. Large steam line breaks would be expected to occur with frequencies no greater than 10^{-3}/yr.	10^{-2}/yr
Branch Probabilities		
Reactor Trip/Demand	A value equivalent to the failure to trip probability developed in WASH- 1400 (3.6 x 10^{-5}/D was employed. Because of the 10^{-6}/yr frequency cutoff used, small variations in the assumed reactor trip probability have no effect on the sequences identified in the study.	3.5x10^{-5}/d
Turbine Trip/ Reactor Trip	All PWR LERs were reviewed for turbine trip, turbine stop valve, etc., failures. While there have been several failures of individual stop valves (single steam lines) to close, only one event (NSIC 092449 at Turkey Point 3, 4) identified a total failure of turbine stop valves. Assuming ~12 shutdowns/plant year (see NUREG/CR-1496) and ~350 PWR years, the number of turbine stop valve demands is ~4200. One failure in this number of demands results in a failure estimate of ~2 x 10^{-4}/Demand.	2 x 10^{-4}/D
Main Feedwater Response/Reactor Trip		
o Failure to Run Back (Overfeed)	The Oconee Pressurized Thermal Shock Evaluation (DPC-RS-1001, January 1982) identifies four failures to run back following a reactor trip: Unit 2, September 17, 1974 Unit 3, April 30, 1978 Unit 2, January 30, 1980 Unit 3, March 14, 1980 During the time period considered in the Oconee study (24 plant years), approximately 144 reactor trips occurred. This results in a failure to run back of 4/144 = 0.03/demand.	0.03/D

Table B.1 (Continued)

Function	Discussion	Value Used
o Failure of MFW Pump Trip Circuitry/MFW Overfeed	This trip circuit is tested yearly. Assuming a general instrumentation failure rate of 10^{-6}/hr (IEEE-500) results in a failure probability of 4 x 10^{-3}/D.	4 x 10^{-3}/D
o Loss of Main Feedwater	NUREG-0560 identifies 5 events at B&W plants up to ~4/5/79 (the date of IE Bulletin 79-05A) in which feedwater was lost subsequent to a reactor trip. If Oconee has a typical reactor trip rate, then the total number of unscheduled reactor trips on these plants up to that date is 6 trips/year x 30.18 years since criticality = 181 trips. This results in a loss of feedwater given reactor trip probability of 5/181 = 0.03. This number does not include instrument bus failures which resulted simultaneously in reactor trip and loss of main feedwater.	
Emergency Feedwater	The emergency feedwater system was assumed to require manual initiation (because of testing or maintenance) approximately 4 hrs/month. This results in a probability of being in manual of 4 hrs/(30 days x 24 hrs/day) = 5.6 x 10^{-3}/D.	5.6 x 10^{-3}/D
EFW Initiated Prior to Steam Generator Dryout		
o EFW Overfeed (One Generator)/Initiator	The EFW level control circuit consists of a level transmitter on each steam generator, an associated E/P converter, and flow control valve. Additional instrumentation is provided to change SG level based on whether or not the RCPs are operating. An alternate level control channel is provided in the event power is lost to the operating channel. It is assumed that accessible instrumentation is tested monthly, and that the level transmitters are effectively tested monthly because level changes during operation can be cross-checked among the S.G. operating and startup range level indications. Based on this, one can roughly estimate the probability of overfill on initiation as the probability of transmitter failure or converter failure of valve failure (~10/hr x 720 hrs (level controllers, IEEE-500) + 1 x 10^{-3}/D (valve failure, WAS-1400)) = 1.7 x 10^{-3}/D.	1.7 x 10^{-3}/D
o EFW Overfeed Not Throttled/Overfeed	It is assumed that the EFW system can be placed in manual and effectively controlled unless the control valve fails open. Valve failures contribute ~60 percent of the EFW overfeed probability developed above. Therefore, the probability of failing to effectively throttle an overfeed was assigned a probability of 0.6/demand.	0.6/D

Table B.1 (Continued)

<table>
<tr><th>Function</th><th>Discussion</th><th>Value Used</th></tr>
<tr><td>o Failure of EFW to Initiate/Demand</td><td>NUREG/CR-2497 identified Emergency Feedwater System failures on demand. This value was 1.1 x 10^{-3}/D including credit for potential rectification. Excluding the possibility of rectification (as would be the case immediately after initiation and prior to any attempt to restart) the observed value was 1.4 x 10^{-3}/D. Oconee observed EFW unavailabilities at the pump level and in 1980 and 1980 include:
<table>
<tr><th>Oconee</th><th>Unavailability</th><th>Date</th></tr>
<tr><td>1</td><td>turbine pump</td><td>5/80</td></tr>
<tr><td>2</td><td>turbine pump</td><td>4/81</td></tr>
<tr><td>3</td><td>turbine pump</td><td>11/80</td></tr>
<tr><td>1</td><td>turbine pump</td><td>6/81</td></tr>
<tr><td>3</td><td>motor driven pump</td><td>5/80</td></tr>
<tr><td>2</td><td>motor driven pump</td><td>3/81</td></tr>
<tr><td>2</td><td>turbine pump</td><td>5/81</td></tr>
</table>
For the 17 month period between 5/80 and 9/81, there have been 7 unavailabilities, which results in an unavailability per pump of 7/(1 test (demand)/mo x 17 mos x 3 pumps/plant x 3 plants) = 0.05/D. With only one pump required for success, one can derive a system failure probability of 0.05 (failure of first pump) x 0.1 (failure of second/first) x 0.3 (failure of third/first and second) = 1.5 x 10^{-3}/D. This is consistent with the NUREG/CR-2497 value of 1.4 x 10^{-3}/D.</td><td>1.4 x 10^{-3}/D</td></tr>
<tr><td>PORV lift/normal MFW or EFW response</td><td>Adequate operational data does not exist to determine the lift probability of the pressurizer PORV during a non-specific reactor trip since the PORV lift setpoint was revised following the TMI-2 accident. Such lifts depend on RCS conditions at the time of the trip as well as potential variations in secondary side response. In lieu of specific operational data, a value of 0.03/demand was assumed.</td><td>0.03/D</td></tr>
<tr><td>Main Feedwater Recovered/Initial Loss of MFW</td><td>A value of 0.5 was assumed. This is consistent with Oconee MFW recovery experience during non-stress situations (EFW initiated) of approximately 0.7/D (P.M. Abrahamson, 10/20/82).</td><td>0.5/D</td></tr>
<tr><td>EFW Initiated After</td><td>As described under "EFW Initiated Prior to Steam Generator Dryout - Failure of EFW to Initiate/Demand," consideration of the potential for short-term rectification following EFW/AFW initiation failures in PWRs results in a decrease in the probability of failing to initiate on demand from 1.4 x 10^{-3}/D to 1.1 x 10^{-3}/D. This is equivalent to a generic probability of initiating EFW/AFW given initial failure of 1 - 1.1 x 10^{-3}/1.4 x 10^{-3} or 0.24/demand.</td><td>0.24/D</td></tr>
</table>

Table B.1 (Continued)

Function	Discussion	Value Used
PORV/PORV Isolation Valve Failure to Close	NUREG-0560 reports 4 failures of the PORV to close when expected once it had opened out of approximately 150 actuations. This results in a failure probability of 4/150 = 0.027 excluding operator action to close the block valve would be expected given the failed-open valve was detected, a probability of failing to terminate PORV flow given the valve was detected open can be calculated as the probability of valve failing to reseat times the probability of failing to close the PORV block valve due to operator error or block valve failure (0.027 x (5x10^{-3} + 1 + 1x10^{-3}) = 1.6x10^{-4}/D). However, because of the closeness of the PORV lift pressure to the safety valve lift pressure, some situations which lift the PORV are expected to also lift the safety valves. Assuming this occurs 10% of the time, that both safety valves will lift if challenged, and that the probability of safety valve failure to reseat is 10^{-2}/demand (WASH-1400), then the additional contribution from these valves is 0.1 x 2 x 0.01 = 2 x 10^{-3}/demand. This results in a total failure to reseat probability of 2.2 x 10^{-3}/D.	2.2 x 10^{-3}/D
Secondary RV Failure to Close on Demand		
o Failure of Any One Valve to Close on Demand	Based on information provided at the 10/20/82 review meeting, Oconee 1-3 has never had an individual failure of these valves to close during a general shutdown (including non-specific reactor trip) or startup. (There have been several multiple failures because of other initiators.) Based on 1978-1980 operating experience, there have been ~10 shutdowns and startups per plant year. Assuming a bypass valve failure on the next shutdown, one can calculate a bounding failure probability as: 1/[no. of observed demands + 1] = 1/[4 valves/plant) x (10 shutdowns + 10 startups/plant year) x (Oconee 1, 2, 3, lifetime through 1981) + 1] = 1/[4 x 20 x 24.2 yrs + 1] = 5 x 10^{-4}/D/valve = 2 x 10^{-3}/D (any of 4 valves)	2 x 10^{-3}/D
o Failure of Any One Valve to Close on Demand (Intact Steam Line during Steam Line Break Mitigation)	The above analysis applies, but only the two valves on the intact steam line need to be considered. This results in a failure probability of 5 x 10^{-4}/D/valve x 2 valves = 1 x 10^{-3}/demand.	1 x 3^{-3}/D

Table B.1 (Continued)

<table>
<tr><th>Function</th><th>Discussion</th><th>Value Used</th></tr>
<tr><td>o Failure of Two Valve on the Same Line to Close on Demand</td><td>While there have been four incidents in which system faults have resulted in turbine bypass valves failing open (Unit 1, 5/4/81; Unit 2, 9/10/74; Unit 3, 5/25/75; and Unit 3, 7/13/75), none of these have occurred during a shutdown sequence. The chance of such a fault occurring following a reactor trip can be bounded as follows: Assuming each reactor trip involves a six hour period which the turbine bypass valves could fail open and that approximately 144 trips have occurred through 1982 (24 plant years x 6 trips/year), the probability of any of the four observed failures occurring following a reactor trip is (4) x (144 trips) x (0.25 days/trip)/(24 plant years) x (365 days/year) = 0.016 fro all trips or ~1 x 10^{-4}/demand.</td><td>1 x 10^{-4}/D</td></tr>
<tr><td>o Failure of Two Valves to Close on Demand (Intact Steam Line During Steam Line Break Mitigation)</td><td>The above analysis applies, but only the valves on the intact steam line need be considered. This reduces the double valve failure value by a factor of 0.5, to 5 x 10^{-5}/demand.</td><td>5 x 5^{-5}/D</td></tr>
<tr><td>Failure to Initiate HPI/Low RCS Pressure</td><td>NUREG/CR-2497 identified HPI system failures on demand. This value was 1.3 x 10^{-3}/D. Oconee observed HPI unavailabilities at the pump/train level between 4/78 and 9/81 (41 plant months) include:
<table>
<tr><th>Oconee</th><th>Unavailability</th><th>Date</th></tr>
<tr><td>3</td><td>train B</td><td>11/78</td></tr>
<tr><td>3</td><td>train A</td><td>4/78</td></tr>
<tr><td>2</td><td>pump A</td><td>2/81</td></tr>
<tr><td>2</td><td>pump B</td><td>1/81</td></tr>
<tr><td>2</td><td>pump B</td><td>11/80</td></tr>
<tr><td>2</td><td>pump B</td><td>7/80</td></tr>
<tr><td>2</td><td>pump B</td><td>7/80</td></tr>
</table>
Considering these seven failures to be unconnected (this is, in actuality, probably not so with the four Oconee 2, pump B failures) and on both a train and pump basis results in the following unavailability estimates:
7/(41 mos x 2 trains/plant x 3 plants = 0.028
and
7/(41 mos x 3 pumps/plant x 3 plants = 0.019
With only one train or pump required or success, one can derive system failure probabilities of
0.028 (first train) x 0.1 (second train/failure of first train) = 2.8 x 10^{-3}/D
and
0.019 (first pump) x 0.1 (second pump/failure of first) x 0.3 (third pump/failure of first and second) = 6 x 10^{-4}/D
These values are consistent with the NUREG/CR-2 value of 1.3 x 10^{-3}/D, which was used.</td><td>1.3 x 10^{-3}/D</td></tr>
</table>

Table B.1 (Continued)

Function	Discussion	Value Used
Failure to Revise SG Level/RCP Trip	The steam generator level setpoint is revised based on signals from the four RCPs when these pumps are tripped. These signals also result in the realignment of feedwater flow to the emergency header. Failure to revise this level was assumed to be dominated by valve failures, and a value of 1×10^{-3}/D was assumed.	1×10^{-3}/D
Failure to Provide SG Cooling Steam Generator Depressurization and Use of Hotwell and Condensate Booster Pumps/Failure of MFW and EFW Systems	A value of 0.1 was assumed in the study. This value only impacts faults on the non-specific reactor trip supplemental response event tree.	0.1/D
Failure to Provide SG Cooling by Further Steam Generator Depressurization and Use of ASW System/Failure of MFW and EFW Systems and Failure to Provide Cooling Using the Hotwell and Condensate Booster Pump	A value of 0.5 was assumed in the study. This value only impacts faults on the non-specific reactor trip supplemental response event tree.	0.5/D

example, for failure to trip the RCPs on HPI injection caused by a turbine bypass valve sticking open after a nonspecific reactor trip or failure to isolate a stuck-open turbine bypass valve under similar circumstances (sequences S14, S15, S17, S31, S32, and S33).

b. For well defined tasks which required specific action and with moderate pressure on the operator, 1×10^{-2}/D was assigned. This value was used, for example, for failure to trip the RCPs after HPI injection during a small-break LOCA. With the exception of the sequences identified in the above paragraph, all sequences were considered "moderate pressure" sequences.

c. For dynamic tasks, such as throttling HPI to maintain prescribed pressure/temperature conditions or reinitiating cooling to an intact steam generator based on pressure and level indications, 3×10^{-2}/D was used.

d. In addition, if a previous operator error had occurred, the applicable error rate for the branch under consideration was doubled. This was the case, for example, for failure to effect secondary isolation on a steam line break, given that the operator had failed to trip the RCPs on HPI initiation.

Two operator response situations included on the initiator event trees are not directly addressed in NUREG/CR-1278; rapid operator response required to mitigate a steam generator overfeed and a plant condition that appeared to be beyond the analysis assumptions for a particular initiator,

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in particular, a failure of a turbine to trip, which would result in the depressurization of both steam generators following secondary isolation. In these cases, the following operator error assumptions were made:

a. For steam generator isolation on main or auxiliary feedwater overfeed (which requires operator response on the order of minutes), a value of 0.1 was used. This value is lower than the NREP screening guide value of ~0.6 for this time period. However, based on the successful manual termination of 10 observed overfeeds at B&W units without main feed pump high level trips (as described in the December 17, 1980 AEOD steam generator overfill observations report), a value of 0.1 is considered defensible.

b. For a situation in which the turbine fails to trip, the probability of an operator's attempting to restore secondary cooling after isolation was assumed to be 0.5.

As discussed previously, branches on the initiator trees dominated by operator action are identified with the symbol 0.

B.4 Thermal-Hydraulic-Related Probability Values

Probability values designated "T/H" on the event trees are estimates dominated by thermal-hydraulic conditions in the plant. These estimates have been developed based on results to date in the Oconee PTS analyses and SAI's experience in prior reactor plant thermal-hydraulic analyses. In general, three values were applied: 0.1 if the state was considered unlikely, 0.9 if the state was considered quite likely, and 0.5 for situations in between theses. Because of the softness in these numbers, sequences involving T/H

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values were not truncated at 10^{-6}/yr unless the sequence frequency, assuming a T/H probability of one, was below 10^{-6}/yr.

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APPENDIX C

ESTIMATION OF PRESSURE, TEMPERATURE, AND HEAT TRANSFER COEFFICIENT

C.1 Introduction

The evaluation of the risks of pressurized thermal shock (PTS) entails the coupling of overcooling incident event trees to fracture mechanics calculation of the probability of vessel crack propagation. The link between an event tree and state and the fracture mechanics calculation is the transient behavior of pressure (P), temperature (T), and heat transfer coefficient (h), in the reactor vessel downcomer region. That is, the P, T, h transient profiles from the sequence defined by an event tree end-state become inputs for the fracture mechanics calculation. There are tens of millions of end-states on overcooling transient event trees. Due to the cost and complexity of thermal hydraulics and fracture mechanics calculations, it is not practical to evaluate every end-state separately. Therefore, it becomes necessary to a) reduce the number of end-states by similarity grouping and b) reduce the number of thermal-hydraulic calculations through the use of less rigorous estimation techniques.

This appendix summarizes the approach used to group sequences and estimate P, T, h profiles for the Oconee-1 PTS study. Section C.2 describes the estimation methodology developed for this study and the approach and rationale for sequence grouping. Sections C.3 through C.8 present the results of evaluations for each of the major initiating events. These include, main steam line breaks (MSLB), turbine bypass valve failures (TBV) at full power and at hot zero power, power-operated relief valve sized primary breaks (PORV LOCA), feedwater transients (FW), and steam generator tube rupture (SGTR). Other initiators such as large-break LOCAs, large main steamline breaks at hot zero power, and small-break LOCAs at hot zero power were not addressed specifically.

The estimates are based on TRAC calculations reported by Los Alamos National Laboratory (J. Ireland, et al., "TRAC Analyses of Severe Overcooling Transients for the Oconee-1 PWR," LA-UR-83-3182, no date), RELAP5 calculations reported by Idaho National Engineering Laboratory (C. D. Fletcher, et. al., "RELAP5 Thermal-Hydraulic Analysis of Pressurized Thermal Shock Sequences for the Oconee-1 Pressurized Water Reactor," EGG-NSMD-6343, July 1983), and plant-specific information available to ORNL. These sources were the data base used in the development of the P, T, h estimates in this appendix. Terminology from these references is used extensively in this appendix. The written descriptions assume reader familiarity with the above named reports as well as availability of the reports for reference to conditions noted in this appendix.

C.2. Methodology

C.2.1 General Approach

After an initial survey of the data resources and the sequences identified for estimation, the five-step process depicted in Figure C.1 was employed in the development of Oconee-1 pressure, temperature, and heat transfer coefficient estimates. This approach allowed logical reduction of the number of cases to be evaluated and derived the greatest benefit from the information in the TRAC and RELAP calculations.

The first step involved the grouping of similar sequences. An evaluation of the TRAC and RELAP calculations for the effects from different operating states provided the criteria for assignment of sequences into groups. Besides providing grouping criteria, step 2 developed the parameters for the cooldown model used on occasion for this study. To assure correct interpretation of conditions during sequences, the appropriate parameters were applied to the cooldown model to duplicate portions of sequences calculated by TRAC or RELAP. This validation effort took place in step 3.

In step 4, the pressure, temperature, and heat transfer coefficients were estimated. Temperature could be estimated either by piecewise application of TRAC and RELAP results or by calculation using the cooldown model. The method selection depended on the complexity of the sequence and the availability or absence of directly applicable data from the TRAC or RELAP calculations. Early portions of sequences where HPI, feedwater, primary loop flows, and downcomer temperatures vary rapidly were not directly applicable to the cooldown model, so piecewise use of TRAC and RELAP

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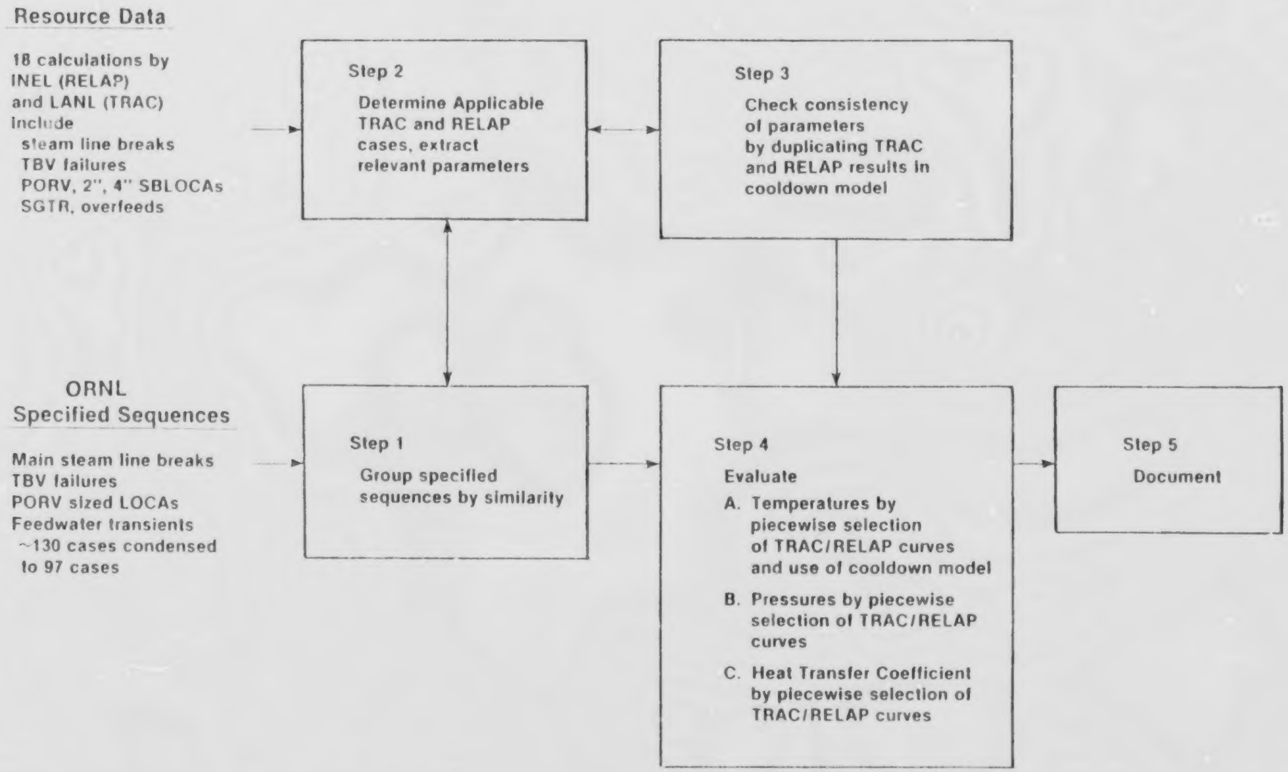
Resource Data
18 calculations by
INEL (RELAP)
and LANL (TRAC)
include
steam line breaks
TBV failures
PORV, 2", 4" SBLOCAs
SGTR, overfeeds
ORNL
Specified Sequences
Main steam line breaks
TBV failures
PORV sized LOCAs
Feedwater transients
~130 cases condensed
to 97 cases
Step 2
Determine Applicable
TRAC and RELAP
cases, extract
relevant parameters
Step 3
Check consistency
of parameters
by duplicating TRAC
and RELAP results in
cooldown model
Step 1
Group specified
sequences by similarity
Step 4
Evaluate
A. Temperatures by
piecewise selection
of TRAC/RELAP curves
and use of cooldown model
B. Pressures by piecewise
selection of TRAC/RELAP
curves
C. Heat Transfer Coefficient
by piecewise selection of
TRAC/RELAP curves
Step 5
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Figure C.1 P, T, h estimation approach.

data was applied. Late in transients where the loop flows are more stable, the cooldown model was applied to predict the temperatures trends. The cooldown model was most useful when the conditions in the requested sequence evaluations deviated significantly from those in the TRAC and RELAP calculations.

Pressure estimates were derived solely from observation of pressure trends in the TRAC and RELAP calculations. The ability of the high head HPI system in the Oconee-1 plant to repressurize the primary quickly was reflected by the TRAC and RELAP5 calculations. The pressure behavior of the various calculations was sufficiently uniform to suggest that piecewise selection of TRAC and RELAP pressure curves was valid.

Heat transfer coefficients were based on piecewise selection of TRAC data. In general, the calculations predicted relatively constant values while the reactor cooling pumps (RCP) are running and step down to a lower but constant value after RCP trip and establishment of natural circulation. The heat transfer coefficients predicted by TRAC did not include correction for free convection effects. Therefore, the values were underpredicted for natural-circulation flow conditions. This effect was compensated for in the fracture mechanics calculations by specifying a minimum heat transfer coefficient of 600 Btu/h $ft^{2o}F$ (3400 W/m^2K). Any lower values reported in this appendix were superceded by the specified limit.

The completed estimations were documented in step 5. This documentation comprises Sections C.3 through C.8 of this appendix.

C.2.2 Sequence Grouping

The overcooling sequence event trees formulated by ORNL possessed over 6.7 million end-states. Application of a 10^{-6}/yr screening limit on event frequency reduced the number of sequences requiring specific estimation to under 200.

Some additional grouping of sequences based on component states' similarities further reduced the number of estimates to 97. However, grouping by component states (i.e., valve sticks open, ICS fails to run back feedwater, etc.) does not account for the thermal-hydraulic impact of such states. The actual impact will also depend on the function or failure of other systems and on feedback from thermal-hydraulic impact of such states. By observation and evaluation of similar events in the TRAC and RELAP calculations, the contributing events in a sequence were classified as dominant, minor, or inconsequential. Sequences with the same dominant events were grouped together for analysis. In later steps, the influence of minor events was evaluated where possible to check the consistency of the groupings. Some sequences were reassigned to other groups based on those checks.

The groupings for each of the initiators are discussed in Sections C.3 through C.8.

C.2.3 Cooldown Model

Most of the overcooling sequences defined by the event tree screening process feature events that vary from the conditions modeled by Idaho

National Engineering Laboratory (INEL) and Los Alamos National Laboratory (LANL). In some instances, the differences are not great, so the TRAC and RELAP5 calculations are representative. In other cases, the differences are very great, so that the calculations are not representative at all. To provide sound estimates of temperature in cases where TRAC and RELAP5 calculated conditions may apply, the cooldown model was developed. The cooldown model is a simple thermal model; no hydraulic calculations are performed. The model requires mass flow rate information that is obtained from TRAC and RELAP5 results and is therefore not a self-contained thermal-hydraulics code. The model is intended to be a simple "first principles" tool to evaluate the impact of heating and cooldown mechanisms on system thermal response with a minimum of time and expense.

The assumptions made to simplify the system are summarized in Table C.1. The assumptions of no heat transfer limitation, thermal equilibrium, and uniform distribution of energy simplify the system to a one-node model, as shown in Figure C.2. The errors due to heat transfer resistance and space-time effects are most noticeable when the system is changing rapidly, i.e., at the beginning of most sequences ($t<1000$ s).

The lumped system in Figure C.2 may be described mathematically as an energy balance:

$$\frac{d(MU)_{sys}}{dt} = Q_D(t) + Q_{RCP} + \dot{m}_{HPI}H_{HPI} + \dot{m}_{FW}H_{FW} - \dot{m}_L H_L - \dot{m}_{ST}H_{ST} \qquad \text{(C.1)}$$

where

$\frac{d(MU)_{sys}}{dt}$ = rate of change in total system energy

Table C.1 Cooldown model assumptions

Assumption	Justifications	Limitations	Model Limits
1. No heat transfer (HT) resistance between primary and secondary	Large HT area Large HT coefficient for boiling, condensation	Loss of heat flow lags and disequilibrium information	Allows lumping of secondary-side and primary-side heat capacities
2. SG secondaries in thermal equilibrium	Assumption 1 Good approximation for choked flow conditions	Not a good approximation where overfeed is compressing steam	Allows use of enthalpy transport model based on choked flow pressure, enthalpy conditions
3. Water inventory is well mixed (energy is uniformly distributed)	Assumption 1 Natural circulation flow is generally much larger than HPI and secondary flows, allowing equilibration or approach thereto	Eliminates space-time effects Difficult to quantify flow stagnation effects	Allows use of 1-node mass-energy balance

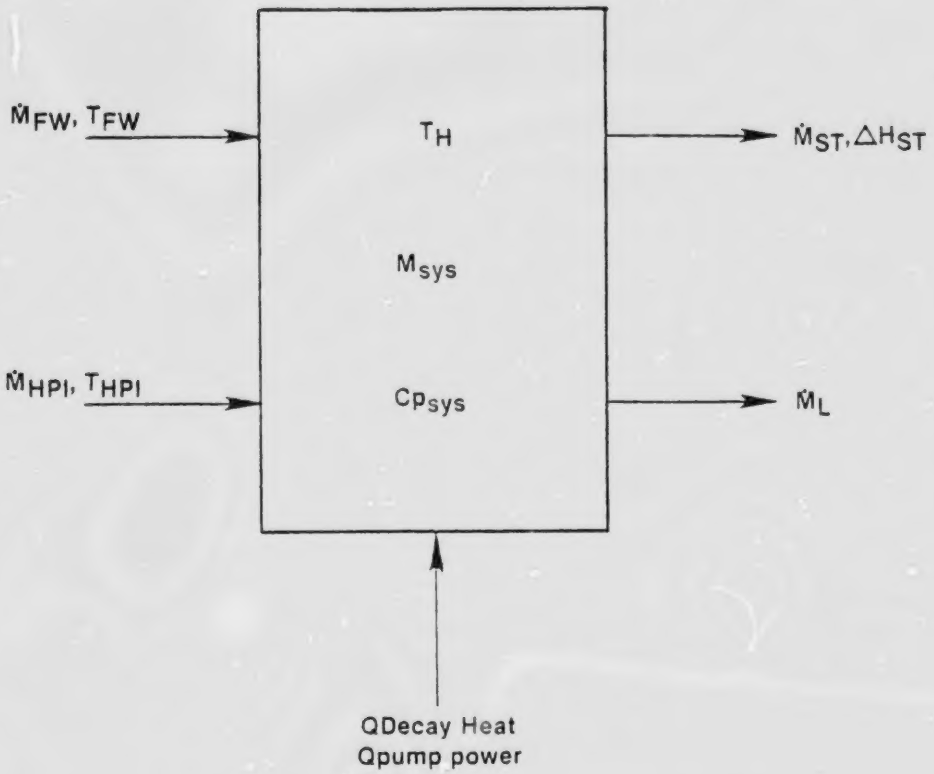
$\dot{M}_{FW}, T_{FW}$
T_H
$\dot{M}_{ST}, \Delta H_{ST}$
M_{sys}
$\dot{M}_{HPI}, T_{HPI}$
Cp_{sys}
$\dot{M}_L$
QDecay Heat
Qpump power

Figure C.2 Inputs and Outputs of system heat balance.

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$Q_D(t)$ = decay heat input as a function of time
(ANS decay heat function)

Q_{RCP} = Pumping power deposited in coolant

$\dot{m}_{HPI}H_{HPI}$ = product of HPI mass flow and specific enthalpy at HPI nominal temperature (T_{HPI}) vs thermodynamic reference temperature (T_{ref})

$= \dot{m}_{HPI}\, C_p\, (T_{HPI} - T_{ref})$

$\dot{m}_{FW}H_{FW}$ = product of feedwater mass flow and specific enthalpy at feedwater temperature (T_{FW})

$= \dot{m}_{FW}\, C_p\, (T_{FW} - T_{ref})$

$\dot{m}_L H_L$ = product of primary leak flow (pressurizer surge line or break) and specific enthalpy of the liquid at the hot temperature (T_H)

$= \dot{m}_L\, C_p\, (T_H - T_{ref})$

$\dot{m}_{ST}H_{ST}$ = product of secondary steam flow and specific enthalpy for satuarated steam at steam generator conditions (T_{SG})

$= \dot{m}_{ST}\, (\Delta H_V\, (T_{SG}) + C_p\, (T_{SG} - T_{ref}))$

The derivative of the total energy may be expanded using the chain rule and simplified as follows.

$$\frac{d(MU)_{sys}}{dt} = M_{sys}\frac{dU}{dt} + U\frac{dM_{sys}}{dt} \qquad (C.2)$$

where

M_{sys} = total system mass

U = specific energy at system temperature, pressure
$= C_v\, (T_H - T_{ref})$

and

$$\frac{dU}{dt} = C_v \frac{dT_H}{dt}$$

$$\frac{dM_{sys}}{dt} = \sum \dot{m} = \dot{m}_{HPI} + \dot{m}_{FW} - \dot{m}_L - \dot{m}_{ST} = \text{net mass crossing system boundary}$$

Substitution into the expansion of the derivative yields

$$\frac{d(MU)_{sys}}{dt} = M_{sys} C_v \frac{dT_H}{dt} + C_v(T_H - T_{ref}) (\dot{m}_{HPI} + \dot{m}_{FW} - \dot{m}_L - \dot{m}_{ST}) \qquad (C.3)$$

For liquids, C_v may be assumed to be equal to C_p. Then substituting the expanded derivative into the energy balance

$$\begin{aligned} M_{sys} C_p \frac{dT_H}{dt} + C_p(T_H-T_{ref}) (\dot{m}_{HPI}+\dot{m}_{FW}-\dot{m}_L-\dot{m}_{ST}) = {} & Q_D(t)+Q_{RCP}+\dot{m}_{HPI} C_P(T_{HPI}-T_{ref}) \\ & + \dot{m}_{FW}C_p(T_{FW}-T_{ref})-\dot{m}_L C_p(T_H-T_{ref}) \\ & - \dot{m}_{ST}(\Delta H_v(T_{SG})+C_p(T_{SG}-T_{ref})) \qquad (C.4) \end{aligned}$$

and then collecting common terms yields

$$\begin{aligned} M_{sys} C_p \frac{dT_H}{dt} = {} & Q_D(t) + Q_{RCP} - \dot{m}_{HPI} C_p(T_H - T_{HPI}) - \dot{m}_{FW} C_p(T_H - T_{FW}) \qquad (C.5) \\ & - \dot{m}_{ST} (\Delta H_v(T_{SG}) + C_p(T_{SG} - T_H)) \end{aligned}$$

Note: $M_{sys} = m(t) = M_{syso} + (\dot{m}_{HPI} - \dot{m}_L + \dot{m}_{FW} - \dot{m}_{ST})(t-t_o)$

This equation gives the change in hot leg temperature for the combined action of the heating/cooldown mechanisms in the model. If all mechanisms except HPI flow are taken to have zero contribution (i.e., are not active), the equation may be solved to yield

$$T_H = T_{HPIin} + (T_o - T_{HPIin}) \exp(\frac{-\dot{m}HPIin}{M_{sys}} t) \qquad (C.6)$$

which is the familiar exponential cooldown model, which form is valid for feedwater overfeeds and HPI injection at zero decay heat.

To model a small steam line break such as an open TBV, it is necessary to specify the steam flow rate, $\dot{m}_{ST}$. Such values are not known a priori. To avoid arbitrary selection of such an important parameter, a relationship between steam flow and hot leg temperature was developed. For isentropic choked flow, the steam flow rate is given by

$$\dot{m}_{ST} = f(P,H)\ A\ P$$

where f(P,H) = mass flow (lb/h/in.^{2} psia (upstream pressure)) as a function of pressure, and mixture enthalpy. See ASME steam tables, fourth edition, Figure 14.

A = break (valve) size (in.^{2})

P = upstream pressure (psia)

By evaluating this expression for saturated steam enthalpy at various pressures and taking a power curve fit against corresponding saturation temperatures, the expression was converted to

$$\dot{m}_{ST} = A \times 1.87045 \times 10^{-11}\ T^{4.32991} \qquad (C.7)$$

which has an accuracy better than ±3% between 200°F and 500°F upstream steam temperature. This steam rate is related to hot leg temperature through an energy balance around the steam generator, which introduces a term for hot leg mass flow (m_H).

$$\dot{m}_{ST}(T_{SG})(\Delta H_V(T_{SG}) + C_p(T_{SG} - T_H)) + \dot{m}_{FW}C_p(T_H - T_{FW}) = \dot{m}_H C_p(T_H - T_{SG}) \qquad (C.8)$$

Given $\dot{m}_H$ and T_H, T_{SG} and $\dot{m}_{ST}$ may be calculated by iteration. The physical meaning of this expression is that the energy transported in by the loop flow (or hot leg mass flow) equals the heat absorbed by the secondary.

With substitution of the equivalent term $\dot{m}_H\ (T_H - T_{SG})$ for the cooldown, Eq. C.8 becomes

$$\frac{dT_H}{dt} = \frac{Q_D(t)}{C_p M_{sys}} + \frac{Q_{RCP}}{C_p M_{sys}} - \frac{\dot{m}_{HPI} C_p (T_H - T_{HPI})}{C_p M_{sys}} - \frac{\dot{m}_H C_p (T_H - T_{SG})}{C_p M_{sys}} \qquad (C.9)$$

Rate of system temperature change	=	heating rate due to decay heat	+	heating rate due to pump power	-	cooldown rate due to HPI flow	-	cooldown rate due to heat transfer to steam generator

where

$$M_{sys} = M_{syso} + (\dot{m}_{HPIin} - \dot{m}_L + \dot{m}_{FW} - \dot{m}_{ST}) (t - t_o)$$

This equation may be solved using Runge-Kutta methods to yield T_H as a function of time. However, important parameters such as $\dot{m}_{HPI}$, $\dot{m}_L$, $\dot{m}_{FW}$, $\dot{m}_H$ and an initial value of T_H are required as input and must be extracted or estimated from the LANL- and INEL-calculated cases.

With T_H and T_{SG} known, downcomer temperature may be calculated by simulating stream mixing around the loop depicted in Figure C.3. Starting at the hot leg, the fluid passes through the steam generator where it is cooled to temperature T_{SG}, as calculated above. The fluid enters the cold leg where it is cooled by mixing with HPI (and is heated by the coolant pumps if running), and it assumes the minimum coolant temperature. On entering the vessel, the coolant mixes with the vent valve flow (at T_H) to yield the downcomer temperature. Vent valve flow is also extracted from the LANL- and INEL-calculated cases. The downcomer temperature is defined by the following equation

$$T_{DC} = \frac{\dot{m}_H T_{SG} + \dot{m}_{HPI} T_{HPI} + \dot{m}_V T_H + (Q_{RCP}/C_p)}{\dot{m}_H + \dot{m}_{HPI} + \dot{m}_V} \qquad (C.10)$$

where

$\dot{m}_v$ = vent valve flow
T_{DC} = downcomer temperature
All other terms are as defined above.

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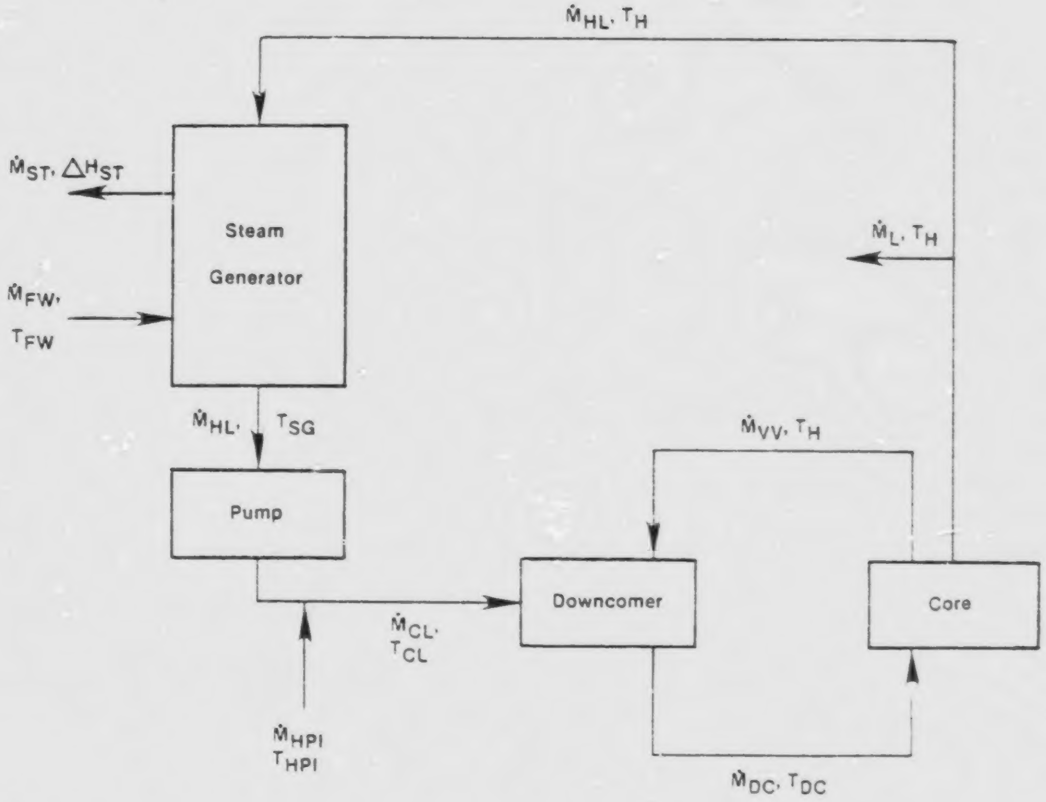
$\dot{M}_{HL}, T_H$
$\dot{M}_{ST}, \Delta H_{ST}$
Steam Generator
$\dot{M}_{FW}, T_{FW}$
$\dot{M}_L, T_H$
$\dot{M}_{HL}, T_{SG}$
$\dot{M}_{VV}, T_H$
Pump
Downcomer
Core
$\dot{M}_{CL}, T_{CL}$
$\dot{M}_{HPI}$ T_{HPI}
$\dot{M}_{DC}, T_{DC}$

Figure C.3 Simple loop model for downcomer temperature evaluation.

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This equation yields mass flow weighted average liquid temperature. Because of the mass flow weighting, the resulting temperature approaches that of the dominating flow, which is the loop flow ($\dot{m}_H$) when natural circulation is strong or the RCPs are running. The downcomer temperature would approach the HPI temperature only if the HPI flow were large relative to the loop and vent valve flow.

C.3 Main Steam Line Break

The main steam line break sequences were the severest cases from a cooldown mechanism standpoint. The cooldown model is not applicable because of the severity. All seven of the ORNL-defined sequences required steam generator isolation after 20 min (1200 s) and restart of the reactor coolant pumps after 22 min (1320 s). For the TRAC and RELAP5 calculations of these events, steam generator isolation was specified at 10 min (600 s), and RCP restart was required on attainment of subcooling margin (300 s for RELAP5 and 526 s for TRAC). The result of these differing specifications is a gap of 800 to 1000 s (between time = 300 and time = 1320 s in the sequences) where there is not sufficient representative data for use in detailed extrapolation. The TRAC and RELAP5 calculations of the main steam line break transient show rapid cooldown trends early in the sequence, prior to restart of the RCPs. The cooldown rate in the downcomer is a complicated function of hot leg temperature primary loop flow, primary-to-steam generator secondary heat transfer (which in turn is a complicated function of pressure, feedwater flow, entrainment losses, etc.), HPI flow, vent valve flow, and downcomer wall-to-coolant heat transfer. All of these parameters will vary to some extent as the transient progresses. The likely response would be that the initial high cooldown rates will not be sustained. In the absence of other means of estimation, the approach used in the evaluation of the main steam line break cases was to use a linear extension of the early cooldown trends. This assumption is clearly conservative. Another conservative assumption covers the use of 212°F as the minimum downcomer temperature. Steam generator boiling heat transfer would dictate this value as a minimum physical limit. HPI and vent valve flow mixing could either raise or lower the temperature from this assumed limit. The main steam line break estimates reported in this section are all subject to these conservative assumptions. The sequences estimated for other initiators are not subject to these assumptions except where noted.

After the main steam line break estimates were incorporated into the fracture mechanics and risk assessment portions of this PTS study, an additional detailed calculation (MSLB parametric case 4) was performed by Los Alamos using TRAC. In this recalculation, the model was changed to properly predict an early trip of the main feedwater pumps and improve measurement of the parameters dictating the manual throttling of HPI and manual restart of one RCP per loop. These changes led to throttling of HPI at 302 s and no restart of the RCPs in parametric case 4 compared to HPI throttling and RCP restart at 526 s for the base case.

These changes result in a declining cooldown rate in the downcomer, suggesting an asymptotic approach to a temperature higher than 212°F. Between the LANL MSLB base case and parametric case 4, sufficient information on cooldown mechanisms could be extracted to allow use of the cooldown model to give a better estimate of the sequences examined in this section. Since the estimates given in this section feature conservatively low downcomer temperatures, revision would probably yield higher temperatures. However, higher minimum temperatures may also mean later steam generator dryout times because heat is retained by the system that would otherwise boil away the water inventory of the broken steam generator. Fracture mechanics calculations would therefore be necessary to assess the impact of P, T, and h revision on vessel failure probability.

The seven sequences defined by ORNL are described in Table C.2 and in the following sections. Table C.3 describes the pertinent features driving the thermohydraulic response of the system. Overspecification exists in some of the cases because other plant protection equipment or thermal-hydraulic phenomena prevent the realization of the specified condition. The basis for the estimation and the estimation procedure are presented, as are tabular data and plots of pressures and temperatures for the sequences.

C.3.1 MSLB 1

C.3.1.1 Basis

Use LANL MSLB base case to determine cooldown and reheating rates.

C.3.1.2 Departures from Basis

a. SG isolation at 20 min instead of 10 min in LANL MSLB base case.

b. HPI throttling and RCP restart at 22 min instead of at attainment of subcooling margin, as in LANL MSLB base case.

C.3.1.3 Applicable Data From Other Cases

a. One-bank TBV failure LANL cases 5A, 5B, 5C, show similar natural circulation d(temp)/d(time) slopes (~0.35°F/s) to that found in the MSLB base case. Further, these cases have no RCP restart and downcomer temperatures that asymptotically approach corresponding loop SG temperatures. Temperatures in SGA decrease, as decreasing steam

Table C.2 Main steam line break study cases

Condition[*]	Case MSLB-1	2	3	4	5	6	7
Secondary Pressure Control							
TSVs close on turbine trip signal	X	X	X	X	X	X	X
TBVs operate as designed	X	X	X	X	X	X	X
SSRVs operate as designed (if challenged)	X	X	X	X	X	X	X
Feedwater							
MFW runback controlled by ICS	X			X			X
MFW overfeed both steam generators			X				
MFW overfeed to broken steam generator		X					
MFW pump trip on high steam generator level (HW pumps, CB pumps continue to operate)		X	X				
MFW pumps trip at time = 0.00					X	X	
EFW initiates on set point	X	X	X	X	X		X
EFW fails to initiate automatically						X	
EFW controlled to level	X	X	X		X		X
EFW overfeed to broken steam generator				X			
Primary Pressure Control							
PORV block valve open	X	X	X	X	X	X	X
PORV operates as designed[**]	X	X	X	X	X	X	
PORV fails to reseat[**]							X
SRVs operated as designed[**]	X	X	X	X	X	X	X

Table C.2 (continued)

	Case MSLB-						
Condition[*]	1	2	3	4	5	6	7
Operator Actions							
RCPs tripped at HPI + 30 s	X	X	X	X	X	X	X
Both SGs isolated at 20 min	X	X	X	X	X		X
Both SGs isolated at 5 min						X	
EFW and TBVs restored to intact SG at 21 min	X	X	X	X	X		X
EFW and TBVs restored to intact SG at 6 min						X	
HPI throttled at later of 22 min or 50°F subcooling	X	X	X	X	X	X	X
1 RCP/loop restarted at later of 22 min or 50°F subcooling	X	X	X	X	X	X	X
Failed PORV isolated when system pressure drops below 2400 psi							X

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[*] Other conditions common to all cases: full power, decay heat 1.0 times ANS, ECCS systems operate as designed

[**] If challenged

Table C.3 P, T, and h profiles for MSLB1, MSLB2, and MSLB7

MSLB1 .T		MSLB1 .P		MSLB1 .h	
Time(s)	T(F)	Time(s)	P(psia)	Time(s)	H(Btu/h) (ft**2F)
0.0	555.0	0.0	2180.0	0.0	2640.0
50.0	487.5	50.0	1000.0	50.0	2640.0
100.0	420.0	100.0	750.0	100.0	285.0
200.0	384.0	200.0	700.0	1320.0	285.0
300.0	348.2	300.0	700.0	1321.0	1160.0
400.0	315.6	360.0	700.0	7200.0	1160.0
500.0	293.0	400.0	750.0		
520.0	286.0	500.0	854.2		
870.0	214.0	520.0	875.0		
1200.0	214.0	1200.0	1582.2		
1320.0	214.0	1320.0	1707.0		
1321.0	296.0	1650.0	1707.0		
1650.0	214.0	3750.0	2430.0		
7200.0	484.0	7200.0	2430.0		

generation rate allows secondary pressure and saturation temperature to drop. The same behavior is assumed in the MSLB sequence.

b. SGB primary flow stagnates in cases 5A, 5B, and 5C, preventing SGB from supplying heat to the primary, which would mitigate downcomer cooldown conditions. Flow stagnation in MSLB extrapolation is assumed to prevail until restart of RCPs (one in each loop).

C.3.1.4 Extrapolation Assumptions and Procedure

a. Temperature -- LANL MSLB base case was used to 520 s with lowest temperature downcomer cell data being interpolated by straight-line segments. Thereafter, extrapolation continued on the same linear trend to 870 s when coldest downcomer region reached 212°F. This lower temperature limit was consistent with nonchoked flow of steam and feedwater out the broken steam line, as observed in both the LANL and INEL calculations. Extrapolation of the loop B void collapse rate prior to 520 s suggests that the voids in the loop B candy cane will also collapse by 870 s. Since loop B could supply heat to the primary after collapse, the effect of collapse will be to limit further temperature decreases.

Downcomer temperature drops to near the cold leg A temperature, which in turn is in equilibrium with the secondary of SGA. After the system cools enough so that entrainment losses decline, SGA begins to fill. EFW flow is controlled on level and ceases prior to isolation at 1200 s. The SGA water inventory of 60,000 lb is boiled out by 1360 s by decay heat, RCP dissipation, and primary heating by SGB. These effects would initially raise downcomer temperature to 296°F because of the mixing of warmer waters from SGB with the downcomer. This effect is also observed in the LANL base case. Steam generator A dryout at 1360 s would result in a downcomer temperature of 220°F. If the steam generator dryout period were extended, the temperatures would remain higher. After 1360 s, the system reheats to 505°F at 7200 s because of decay heat and RCP power input. Table C.3 and Figure C.4 summarize the results for this case.

b. Pressure -- MSLB base case data were used to 520 s. The system pressure drops to 700 psia by 200 s. HPI begins to offset coolant shrinkage by 400 s and increases pressure by 1.04 psi/s between 400 and 520 s. This rate is assumed to continue until HPI throttling at 1320 s (22 min), after which pressure is assumed constant at 1707 psia (may actually drop) between 1320 s and SGA dryout at 1360 s. Then repressurization occurs at a rate of 0.3446 psi/s to the PORV set point, 2430 psia at 3460 s, following which the pressure remains constant until the end of the sequence (7200 s). Table C.3 and Figure C.4 summarize the results for this case.

c. Heat Transfer Coefficient -- Heat transfer coefficients for the MSLB TRAC base case were obtained from Appendix A of the LANL Oconee-1 report. The initial value of 2640 Btu/h ft^{2}°F (15000 w/m^{2}k) persists until the RCPs trip at ~50 s. The heat transfer coefficient then ramps down with loop flow to 285 Btu/h ft^{2}°F (1620 w/m^{2}k) at 100 s and is assumed to remain at this value until restart of two RCPs at 1320 s. This value neglects decreases due to reduction of natural circulation

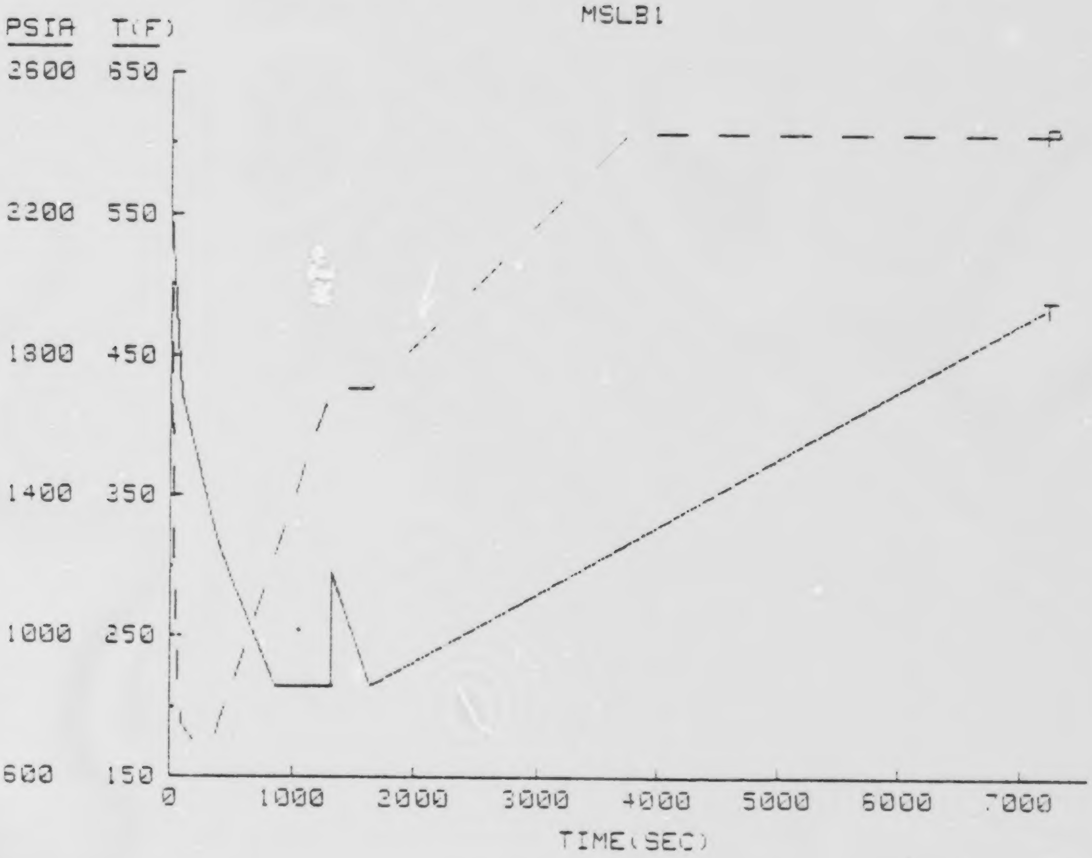

Figure C.4 Pressures and temperatures for MSLB1, MSLB2, and MSLB7.

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flows and elimination of vent valve flows. After RCP restart, a value of 1160 Btu/h ft$^{2\circ}$F (6600 W/m^{2}K) was obtained for the remainder of the transient. The results are summarized in Table C.3.

C.3.2 MSLB2

C.3.2.1 Basis

LANL MSLB base case to determine cooldown and reheating rates.

C.3.2.2 Departure from Basis

a. MFW overfeed to SGA with MFW trip on level.

b. EFW controls to level.

C.3.2.3 Applicable Data From Other Cases

LANL MSLB case 3 features MFW pump failures to trip, but the result is overfill of SGB, which trips MFW. Likewise, the MSLB base case suffers a MFW pump trip (low suction P) when the ICS is running back the feedwater. An overfeed where there is no valve runback would result in MFW pump trip under these conditions or malfunction if it remained on.

C.3.2.4 Extrapolation Assumptions, Procedures, and Results

a. Temperature -- At the time of MFW trip (45 s) in the MSLB base case, the MFW temperature has started dropping rapidly, indicating flashing to saturation conditions in the feedwater lines. Heat loss through SGA would be limited by natural circulation to approximately 370 lb/s equivalent in steam with an additional amount of steam lost

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in feedwater flashing. With a nominal SGA feedwater rate of 1500 lb/s and assuming no liquid losses to entrainment from the break, a net inflow of 1230 lb/s would fill the steam generator to 120,000 lb within 100 s after RCP trip. Downcomer temperature response would show an early approach to 212°F but probably would not drop below that value because of vent valve flow. The case may be approximated by MSLB1 and MSLB4. These cases provide the bounds for SGA dryout time although downcomer temperatures may be lower between 50-870 s. MSLB1 is a reasonable approximation because the accumulated decay heat and primary heat inventory would be capable of evaporating the excess water (>60,000 lb) in SGA plus an additional 100-200 lb/s of EFW and therefore would be at the same state as MSLB1 at isolation (1200 s). Table C.3 and Figure C.4 summarize the results for the roughly equivalent MSLB1 case.

b. Pressure -- Pressures for portions of the transient will be less than those for file MSLB1. Use of MSLB1 would therefore be conservative.

c. Heat Transfer Coefficient -- Used same values for file MSLB1.H.

C.3.3 MSLB3

C.3.3.1 Basis

LANL MSLB parametric case 3 (steam line break with runaway feedwater).

C.3.3.2 Departures from Basis

a. EFW actuates and controls on level.

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b. Isolation of SGA at 1200.

c. Restart 1 RCP/loop and throttle HP1 at 1320 s.

C.3.3.3 Applicable Data from Other Cases

In LANL MSLB base case, total EFW and HWP/CBP flow rate to SGA is 400 lb/s; one-bank TBV failure cases LANL 5A, 5B, 5C show downcomer temperatures; asymptoticaly approaching respective SG temperatures.

C.3.3.4 Extrapolation Assumptions, Procedures and Results

a. Temperatures -- In MSLB parametric case 3, the MFW pumps do not trip until high level in SGB is encountered at 331 s. Therefore, the minimum downcomer temperature profile for parametric case 3 was employed to 330 s. A temperature of 310°F is obtained at this point. Prior to MFW pump trip, a feedwater flow rate of approximately 440 lb/s to SGA is obtained. The flow rate afterwards is 330 lb/s. These values compare with 400 lb/s for EFW and HW/CB pump total flow rate for the LANL MSLB base case. Pump head curves and flow resistance in the EFW spray ring may account for the similarities. The effect of feedwater entrainment losses from SGA could not be properly evaluated. For extrapolation purposes, an EFW-HW/CB pump total flow of 440 lb/s was assumed. Therefore, the cooling trend is continued until the A-loop cold legs reach 212°F at 904 s. This temperature will persist until restart of the RCPs at 1320 s. Water starts to accumulate in SGA due to the natural circulation mass flow limitation on heat transfer. Approximately 60,000 lb of water will be on hand in SGA at isolation. At the restart of the RCPs (1320 s), downcomer temperature will jump to the primary average, 270°F, and drop to 228°F at 1350 s as the SGA secondary dries out. Thereafter, the system reheats at 0.0489°F/s until 7200 s when it reaches 514°F (see Table C.4 and Figure C.5).

b. Pressure -- System pressure drops to 800 psi at 125 s and then increases at 0.77 psi/s until HPI throttling at 1320 s. Pressure is 1722 psi at that point. Thereafter, system reheat increases pressure by 0.3446 psi/s until the PORV set point is reached at 3375 s and remains constant through 7200 s (see Table C.4 and Figure C.5).

c. Heat Transfer Coefficient -- While RCPs are running, the heat transfer coefficients are the same as for corresponding portions of MSLB1. The similar natural circulation rates of MSLB1 will cause similar values of h. Therefore, the h values predicted for MSLB1 apply to this case also (see Table C.4).

C.3.4 MSLB4

C.3.4.1 Basis

LANL MSLB base case.

C.3.4.2 Departures from Basis

Same as MSLB1 except EFW level control fails, leading to overfeed of SGA.

C.3.4.3 Applicable Data From Other Cases

See MSLB1.

C.3.4.4 Extrapolation Assumptions and Procedures

Table C.4 P, T, and h profiles for MSLB3

MSLB3 .T		MSLB3 .P		MSLB3 .h	
Time(s)	T(F)	Time(s)	P(psia)	Time(s)	H(Btu/h)(ft**2F)
0.0	555.0	0.0	2180.0	0.0	2640.0
50.0	477.0	50.0	1000.0	50.0	2640.0
330.0	390.0	125.0	800.0	100.0	285.0
904.0	212.0	1320.0	1722.0	1320.0	285.0
1320.0	212.0	3375.0	2430.0	1321.0	1160.0
1321.0	270.0	7200.0	2430.0	7200.0	1160.0
1350.0	228.0				
7200.0	514.0				

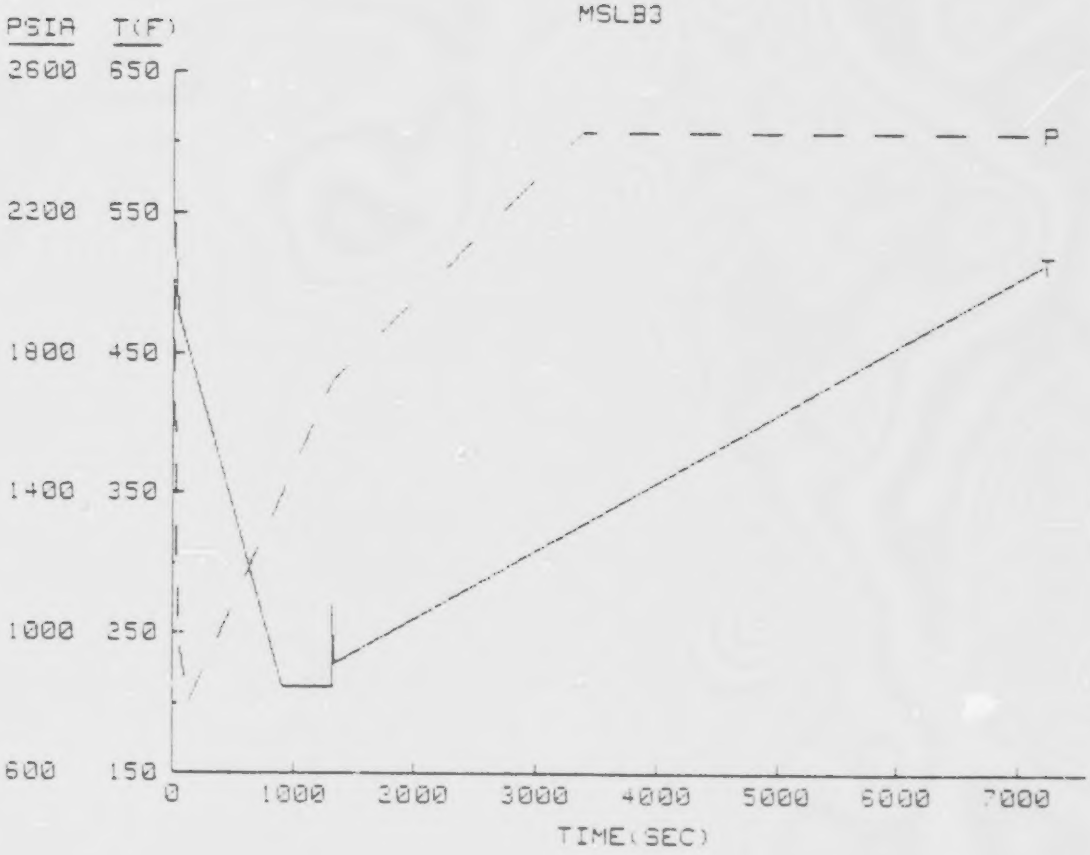

Figure C.5 Pressures and temperatures for MSLB3.

a. Temperature -- Case is the same as MSLB1 to 1320 s. Minimum temperature of 212°F is reached at 870 s, as in MSLB1. Entrainment losses could not be evaluated and were conservatively ignored. SGA and steam line A water inventory at isolation (1200 s) is 150,000 lb compared to 60,000 lb for MSLB1 at same instant. Natural circulation flows limited cooldown rate prior to isolation. After RCP restart at 1320 s, downcomer temperature rises to 296°F due to mixing of downcomer and SGB waters, as described in MSLB1. Downcomer temperature returns to 212°F at 1380 s as SGB and primary system heat inventories are exhausted, boiling 70,000 lb of water from SGA in the process. Decay heat and RCP heat input (12725 Btu/s for two pumps) boil dry the remaining water (80,000 lb) by 4090 s. System heat at 0.0489°F/s returns the system to 364°F at 7200 s (see Table C.5 and Figure C.6).

b. Pressure -- Pressure behaves similarly to MSLB1 to 1320 s, that is, depressurization to 700 psi by 200 s, repressurization to 1707 psi at 1320 s. Pressure is assumed to hold constant (would probably drop somewhat instead) at 1707 psia until SGA dryout at 4010 s, since HPI has been throttled and there is no system reheat during this period. Afterwards, pressure climbs at .33446 psi/s to PORV set point (2430 psia) at 6190 s and remains until 7200 s (see Table C.5 and Figure C.6).

c. Heat Transfer Coefficient -- Case is identical to MSLB1 (see Table C.5).

Table C.5 P, T, and h profiles for MSLB4

MSLB4 .T		MSLB4 .P		MSLB4 .h	
Time(s)	T(F)	Time(s)	P(psia)	Time(s)	H(Btu/h) (ft**2F)
50.0	487.5	50.0	1000.0	50.0	2640.0
100.0	420.0	100.0	750.0	100.0	285.0
200.0	384.0	200.0	700.0	1320.0	285.0
300.0	348.2	300.0	700.0	1321.0	1160.0
400.0	315.6	360.0	700.0	7200.0	1160.0
500.0	293.0	400.0	750.0		
520.0	286.0	500.0	854.2		
870.0	212.0	520.0	875.0		
1200.0	212.0	1200.0	1582.2		
1320.0	212.0	1320.0	1707.0		
1321.0	296.0	3430.0	1707.0		
1380.0	212.0	5530.0	2430.0		
4090.0	212.0	7200.0	2430.0		
7200.0	364.0				

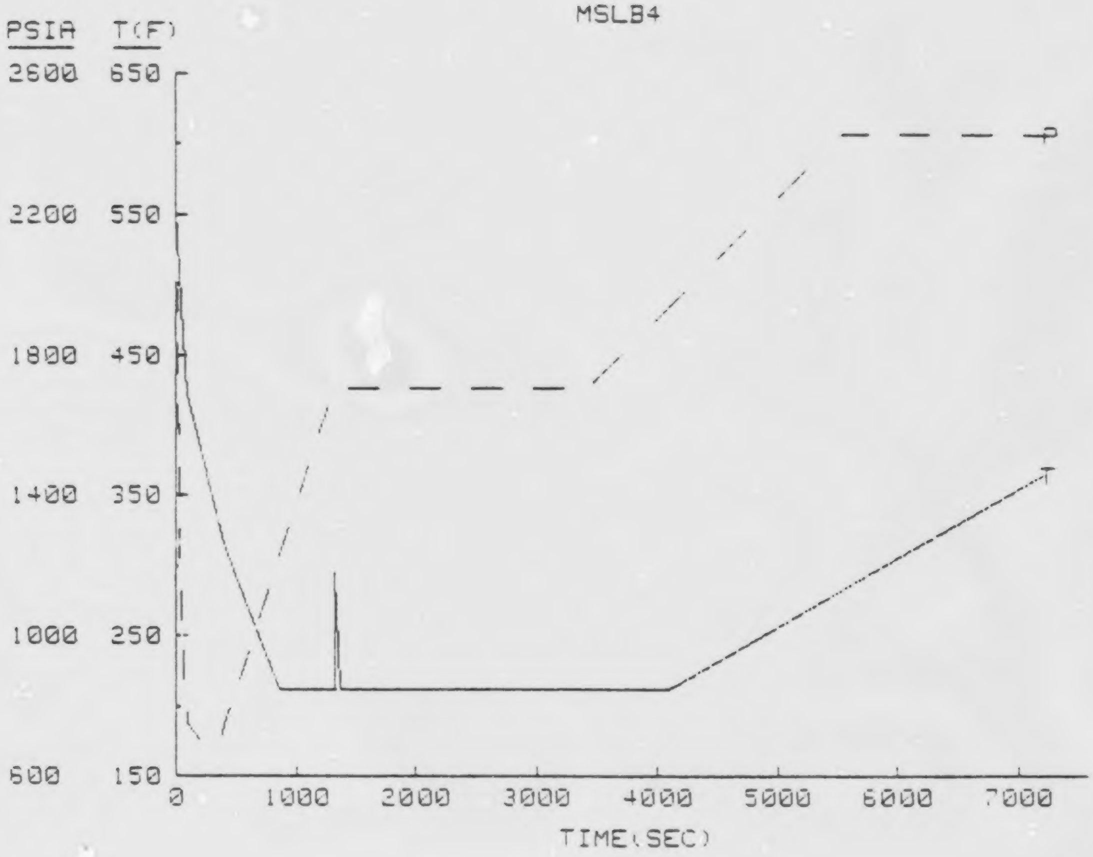

Figure C.6 Pressures and temperatures for MSLB4.

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C.3.5 MSLB5

C.3.5.1 Basis

LANL MSLB base case.

C.3.5.2 Departures from Basis

Main feedwater pumps trip at time 0.0. HW/CB pumps and EFW function properly.

C.3.5.3 Applicable Data from Other Cases

See MSLB1.

C.3.5.4 Extrapolation Assumptions, Procedures, and Results

a. Temperature -- Loss of MFW at time of break would reduce water available for boiling in SGA over first 50 s of transient from 71,500 lb to only 26,000 lb. As a result, temperature drops to 492°F rather than 487°F in MSLB1. Thereafter, the same cooldown slopes from MSLB1 are used to project temperatures at 350 s (388°F) and 1150 s (212°F). An accumulated inventory of 60,000 lb of water in SGA is expected at isolation. From 1320 s on, the behavior is taken to be the same as MSLB3 (see Table C.6 and Figure C.7).

b. Pressure -- Since the initial cooloff will be less severe than the base case, depressurization to the HPI set point would be slightly delayed. The heat contribution of SGB to the primary is expected to sustain primary pressure, as occurred in MSLB3. In the absence of any reasonable way to predict the delay or extent of depressurization,

Table C.6 P, T, and h profiles for MSLB5

MSLB5 .T		MSLB5 .P		MSLB5 .h	
Time(s)	T(F)	Time(s)	P(psia)	Time(s)	H(Btu/h) (ft**2F)
0.0	555.0	0.0	2180.0	0.0	2640.0
50.0	493.0	50.0	1000.0	50.0	2640.0
350.0	388.0	125.0	800.0	100.0	285.0
1150.0	212.0	1320.0	1722.0	1320.0	285.0
1320.0	212.0	3375.0	2430.0	1321.0	1160.0
1321.0	270.0	7200.0	2430.0	7200.0	1160.0
1350.0	228.0				
7200.0	514.0				

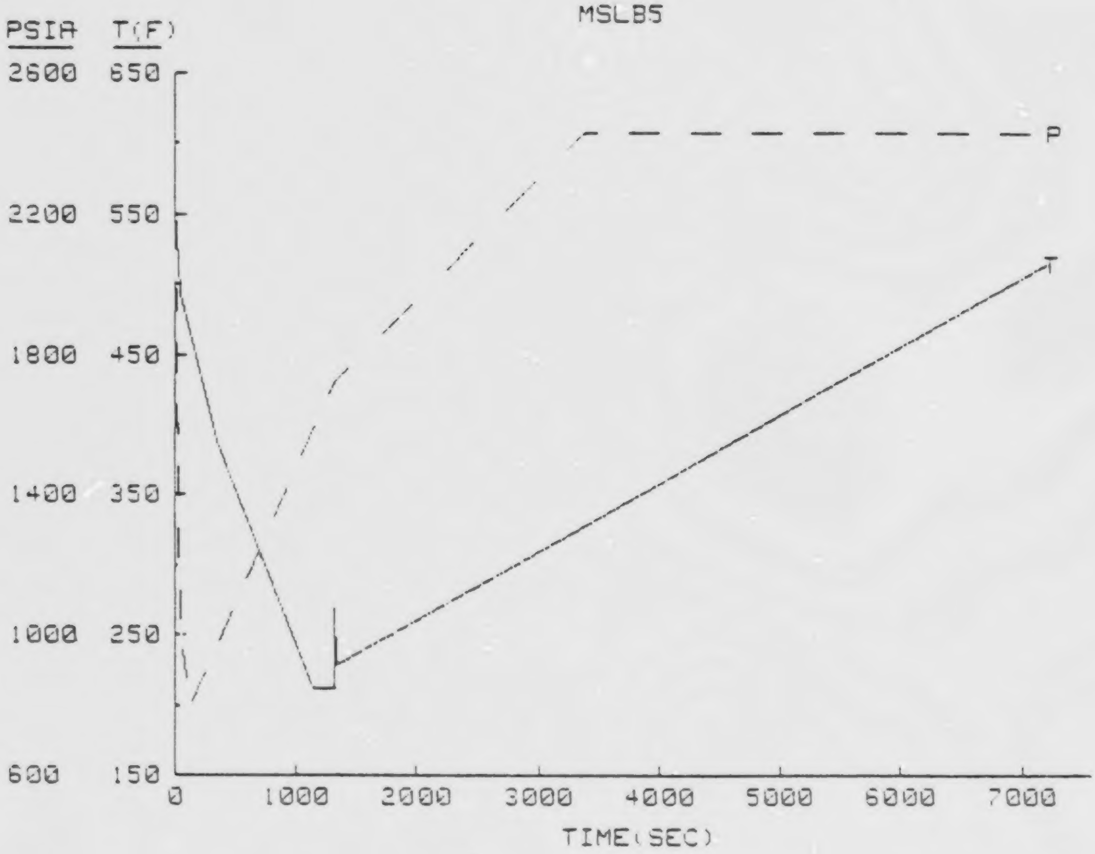

Figure C.7 Pressures and temperatures for MSLB5.

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the use of the pressure curve for MSLB3 would represent a conservative estimate (see Table C.6 and Figure C.7).

c. Heat Transfer Coefficient -- Case is identical to MSLB1 (see Table C.6).

C.3.6 MSLB6

C.3.6.1 Basis

LANL MSLB base case for initial response of system for transient, LANL Rancho Seco-type transient for response of system to refill of dry SGB.

C.3.6.2 Departures From Base Cases

a. Loss of all main feedwater flow time 0.0.

b. EFW in manual control.

c. SGs isolated at 5 min.

d. SGB restored at 6 min with full EFW flow.

C.3.6.3 Applicable Data from Other Sources

Not applicable.

C.3.6.4 Extrapolation Assumptions, Procedure, and Results

a. Temperature -- Along with the loss of main feedwater at the beginning of the sequence, it is assumed that the hotwell/condensate booster pumps are also unable to deliver flow to the affected steam generator. As a result, SGA dries out very quickly (probably less than 20 s)

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because of the liquid entrainment during the initial blowdown. From the LANL MSLB calculation, the loop A downcomer region reaches 510°F at this moment and the loop B downcomer remains near 570°F. Having lost all of its water inventory, SGA no longer contributes to system cooldown, and the downcomer temperature soon returns to the hot leg temperature, > 570°F. SGB dries out by about 50 s into the transient. The cooldown model was used to extrapolate a system temperature rise to 586°F by 6 min (360 s) when EFW is restored to SGB. With EFW restricted to SGB only and the TBVs on SGB controlling to pressure, the system (hot leg) temperature slowly decreases to 548°F. Downcomer temperature decreases to 534°F at 460 s because of mixing of cold HPI with ever-reducing amounts of loop flow, more than offsetting system heatup. However, as HPI flow decreases as the system reaches PORV set point pressure at 200 s, some mixing effects and increased loop flow to HPI flow ratio cause the downcomer temperature to rise to 546°F and to continue to rise to 556°F at initiation of EFW at 360 s. Assuming a natural circulation flow of 900 lb/s in loop B, maintenance of the SG exit temperature at 548°F, 70 lb/s of HPI flow, and a vent valve flow of 45 lb/s, the average downcomer temperature would fall from 556°F at 360 s to 514°F before 460 s because of EFW initiation. The cooldown is not as dramatic as in the Rancho Seco-type of event because, in this case, only one SG is involved and EFW is controlled to level. After the SG fills to level, the downcomer temperature holds steady at around 515°F because of the TBV operation on SGB and continued HPI flow. At 1200 s the HPI is throttled and one RCP per loop is restarted. This results in excellent mixing in the downcomer with a temperature equal to the mean of the hot leg and SGA exit temperatures. Downcomer temperature rises to 564°F soon after RCP restart and decays to 548°F by 1420 s, after which SGB regulation keeps the system at 548°F for the remainder of the sequence (see Table C.7 and Figure C.8).

b. Pressure -- The combination of main steam line break and reactor trip causes HPI initiation (set point 1500 psia) at 20 s. However, the prompt boil-dry of SGA because of the break and SGB because of loss of both MFW and EFW causes rapid system reheat, which, along with HPI flow, results in an equally rapid repressurization. The PORV set point pressure, 2430 psia, is reached at 200 s (compare with 226 s for the Rancho Seco-type sequence). Pressure decreases upon restart of the RCPs because of enhanced cooldown and coolant shrinkage. LANL cases 5A and 5B calculated pressure reductions on the order of 500 psi on RCP restart; a similar trend was assumed here. After reaching 1900 psia at 1300 s, the pressure is assumed to increase at 0.34 psi/s, as in the LANL MSLB calculation. However, pressure recovery may not be significant because the system temperatures are near the hot standby level so that reheating and expansion will be minimal (see Table C.7 and Figure C.8).

c. Heat Transfer Coefficient -- Heat transfer coefficient drops from 2820 Btu/h ft^{2}°F at 50 s to 285 Btu/h ft^{2}°F at 100 s because of trip and spindown of the RCPs. The natural circulation value persists until restart of one RCP per loop at 1200 s, whereon the value climbs to 1160 Btu/h ft^{2}°F for the remainder of the sequence (see Table C.7).

Table C.7 P, T, and h profiles for MSLB6

MSLB6 .T		MSLB6 .P		MSLB6 .h	
Time(s)	T(F)	Time(s)	P(psia)	Time(s)	H(Btu/h) (ft**2F)
0.0	555.0	0.0	2180.0	0.0	2640.0
20.0	510.0	20.0	1500.0	50.0	2640.0
50.0	566.0	200.0	2430.0	100.0	285.0
160.0	534.0	1200.0	2430.0	1200.0	285.0
200.0	546.0	1300.0	1900.0	1201.0	1160.0
260.0	550.0	2850.0	2430.0	7200.0	1160.0
360.0	556.1	7200.0	2430.0		
460.0	515.0				
600.0	514.0				
1200.0	514.0				
1201.0	564.0				
1240.0	561.0				
1300.0	555.0				
1360.0	550.0				
1400.0	548.0				
7200.0	548.0				

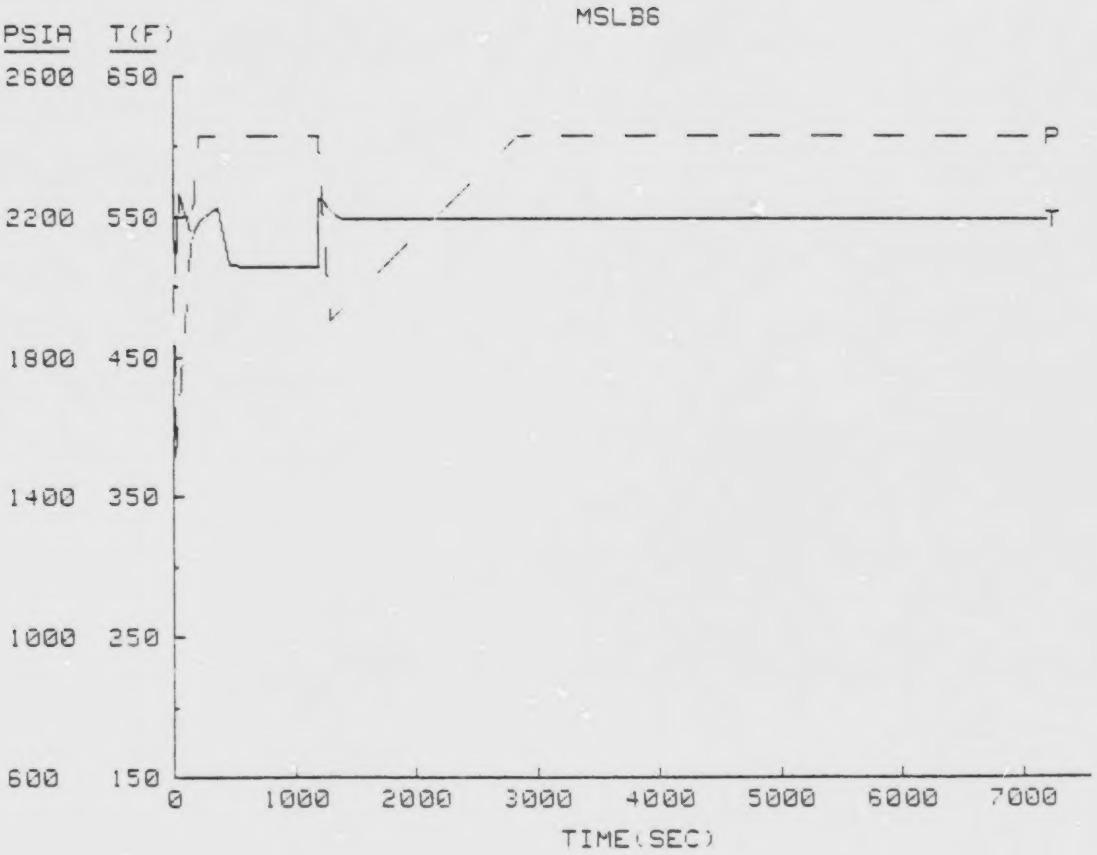

Figure C.8 Pressures and temperatures for MSLB6.

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C.3.7 MSLB7

This case features the same sequence as MSLB1 except that the PORV is assumed to fail open when challenged and is isolated when primary pressure drops back below 2400 psi. In MSLB1, the PORV is not challenged until 3460 s and there is no major heat source to keep the system pressure up. Therefore, the block valve could be closed promptly. This case would then be a subset of MSLB1 (see Table C.3 and Figure C.4).

C.4 Turbine Bypass Failures at Full Power

C.4.1 ORNL-Defined TBV Cases (19 Total)

ORNL defined 19 turbine bypass valve failure cases for full power operations. These were sorted into 10 groups designed to have similar transient responses, especially in the later stages when temperatures are lowest. Table C.8 describes the characteristics of the sequences and lists group assignments. The transients are evaluated by groups below.

C.4.1.1 Base

TBV failures in one loop -- LANL 5A, 5B, 5C.

TBV failures in two loops -- LANL 6A, 6B, 6C.

C.4.1.2 Departures from Bases

a. All study cases fail, only one TBV in a bank. LANL cases fail, both TBVs in a bank.

b. All study cases have EFW operate as designed and control SG inventory to level. All LANL cases feature runaway feedwater.

c. The study cases feature a variety of MFW behaviors (overfeeds to one or both SGs, normal runback, and trips coincident with turbine trip and/or reactor trip). The LANL cases all feature overfeeds to SGA or both SGs.

Table C.8 Full power TBV failure cases groupings

Condition*	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19
	Case (TBV-)																		
Secondary Pressure Controls																			
1 TBV open SGA	x	x	x	x	x	x	x		x	x			x	x	x				
1 TBV open SGA, SGB								x								x	x	x	x
1 TBV open SGA, 1 SRV open SGA											x	x							
Feedwater																			
MFW runback as designed	x		x					x	x	x	x	x	x					x	x
MFW overfeed SGA, high level trip				x										x			x		
MFW overfeed both SGs, HL trip						x	x												
MFWP trip at time = 0.0		x			x										x	x			
EFW control to level	x	x	x	x	x	x	x	x	x	x	x	x	x	x	x	x	x	x	x

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Table C.8 (continued)

	Case (TBV-)																		
Condition*	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19
Primary Pressure Controls																			
PORV block valve closed									x										
PORV operates properly	x	x	x	x	x	x	x	x			x	x		x	x	x	x		
PORF fails to reseat**										x			x					x	
SRV fails to reseat**													x	x	x				x
Operator Actions																			
RCP tripped at HPI + 30 s	x	x	x	x	x	x	x	x	x	x	x	x	x	x	x	x	x	x	x
No other actions	x	x					x	x	x		x	x	x	x	x	x	x	x	x
SG isolation at 20 min, SGB restored at 21 min			x	x	x	x				x									
HPI throttled later of 22 min or 50°F subcooling			x	x	x	x				x									
1 RCP/loop restart later of 22 min or 50°F subcooling			x	x	x	x				x									
Group number (TBVG-)	1	1	2	3	2	3	4	5	1	6	8	7	6	4	6	5	9	10	10

*Other conditions common to all cases: full power, decay heat 1.0 times ANS, all ECCS components operate as designed

**If challenged

C.4.1.3 Applicable Data from Other Cases

Early behavior of system with MFW trip at time = 0.0 is available from LANL Rancho Seco-type transient. The LANL and INEL cases on PORV failures and PORV-sized hot leg LOCAs are of some limited use for the study cases including such failure. The limitations stem from water-solid conditions in the study cases compared to normal pressurizer conditions at the start of the LANL and INEL cases.

C.4.1.4 Extrapolation Assumptions

a. Temperature

One-Loop TBV Failure--Application of the cooldown model to a one loop TBV failure requires knowledge of natural circulation and vent valve flow rates and of the effects of HPI injection on stagnated loop cold leg temperature. LANL cases 5A and 5B were examined with regard to these requirements. The resulting data and assumptions are: (1). Loop A natural circulation remains steady at about 1023 lb/s from 400 s until 1500 s and is assumed to continue until 7200 s. Loop B stagnates at about 600 s and is assumed to remain so until 7200 s. The stagnation of loop B is due to SGBs being warmer than the primary. The density differences and corresponding buoyancy effects from higher loop temperature cancel the forces in the vessel that would drive circulation in the loop. (2). Vent valve mass flow rates corresponding to 264 lb/s (120 kg/s) commence at 600 s and hold steady through 1500 s and are assumed to continue to 7200 s (valid as long as SGB remains stagnant). (3). Unthrottled HPI flows total 40 lb/s (18 Kg/s) in loop A and 30 lb/s (14 kg/s) in loop B at the PORV set

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point pressure (2450 psia). Loop A HPI flow thoroughly mixes in the cold leg. Loop B HPI is not mixed because of stagnation. At 1500 s the temperature is 134°F at the loop B HPI injection points and 207°F at the loop B cold leg nozzles at the vessel. Both temperatures are declining at a rate of 0.23 °F/s because continued HPI flow mixing with the coolant in the cold legs, the gradual displacement of water from the cold legs, and fluid heating by the pipe walls of the cold legs. If these trends continue, 50°F (i.e., the HPI injection temperature) water would be encountered at 1860 s and 2175 s at these respective locations of loop B. (4). The turbine bypass valve area of 0.16 ft^2 as taken from the LANL model was halved in the calculations to simulate failure of only one TBV in the bank. (5). In some cases, a quasi-static temperature was calculated at 1500 s to provide a point for linear extrapolation to a region where the cooldown model could be easily applied. The quasi-static temperature calculation assumes that the energy loss rate in the steam leaving the stuck-open TBV equals the decay heat rate. A unique steam generator temperature is obtained for this quasi-static energy balance. The downcomer temperature is obtained from a mixing calculation considering this steam generator condition and HPI and vent valve flow rates and temperatures.

Two-Loop TBV Failure--To apply the cooldown model to the two-loop TBV failure, the following information from LANL case 6A, 6B and assumptions are required: (1). Balanced loop natural circulation flows of 842 lb/s for each loop are assumed to persist to 7200 s. (2). Vent valve velocities after 250 s approach values corresponding to 44 lb/s (20 kg/s). It is assumed that this value holds to 7200s. (3). Unthrottled HPI flows total 40 lb/s (18 kg/s) in loop A and 30 lb/s (14 kg/s) in loop B at the PORV set point pressure. (4). The turbine bypass valve area of 0.16 ft^2 from the LANL model was halved in the calculations to simulate failure of only one TBV per loop. (5). Where applicable, the quasi-static temperature (as discussed above) is used for extrapolation of the initial 1500 s of a sequence.

b. Pressure

Pressure is determined on a case-by-case basis and will depend on intial cooldown mechanism. Repressurization rates observed in LANL cases 5A, 5B, and 5C and 6A, 6B, and 6C are applied as engineering judgment suggests.

c. Heat Transfer Coefficient

Heat transfer coefficients are primarily a function of fluid velocity. Four regimes exit in the study cases: four RCPs operating, natural circulation (two-loop TBV failure), natural circulation with one loop stagnant (one-loop TBV failure), and two RCPs (one per loop) operating. LANL estimated heat transfer coefficients of 16000 W/m^2K (2820 Btu/h $ft^{2o}F$) for four-RCP operation, 1200 W/m^2K (210 Btu/h $ft^{2o}F$) for both natural circulation cases, and 7500 W/m^2K (1320 Btu/h $ft^{2o}F$) for the vessel cells receiving full flow (also the coolest water) from the operating RCP in 1 RCP/loop operation. These values are applied as the transient sequence dictates.

C.4.2 Group 1 -- Cases TBV1, TBV2, and TBV9

a. Temperature

PTS concerns will be most evident late in these transients. Main feedwater disturbances early in these sequences will not greatly influence final temperature although early cooldown slopes will be different. The temperature curve (file TBVG1.T) was constructed using the first 60 s of LANL case 5B (valid to that point), straight-line extrapolation to the quasi-static temperature of 366°F at 1500 s, and followed by extrapolating with the cooldown model to 297°F at 7200 s (see Table C.9 and Figure C.9).

Because of stagnated conditions in loop B, local temperatures may be depressed because of the absence of other flow to mix with HPI. This effect was estimated using data from the cooldown model to calculate the temperature obtained from mixing the loop B HPI flow (30 lb/s) and one-half of the total vent valve flow (132 lb/s). A minimum temperature of 270°F was obtained at 7200 s. This minimum should be reexamined should vent valve flow be reduced, as much lower temperatures would result (see Table C.10 and Figure C.10)

b. Pressure

Pressure was taken to follow the same profile as LANL case 5B. This is conservative, since the lower cooldown rates will extend the period of time before HPI comes on and reduce the time for HPI to recover system pressure to the PORV set point, 2430 psi. As modeled, the pressure drops to 1500 psia at 150 s and remains there until 320 s, after which the pressure rises to its final value, 2430 psi at 975 s. Case TBV9 will exhibit

Table C.9 Nominal P, T, and h profiles for TBV group 1

TBVG1 .T		TBVG1 .P		TBVG1 .h	
Time(s)	T(F)	Time(s)	P(psia)	Time(s)	H(Btu/h) (ft**2F)
0.0	555.0	0.0	2180.0	0.0	2820.0
5.0	572.0	60.0	1740.0	183.0	2820.0
60.0	552.0	150.0	1500.0	233.0	210.0
1500.0	366.0	320.0	1500.0	7200.0	210.0
2500.0	352.6	710.0	2000.0		
3500.0	339.8	975.0	2430.0		
4500.0	327.6	7200.0	2430.0		
5500.0	315.8				
6500.0	304.6				
7200.0	297.0				

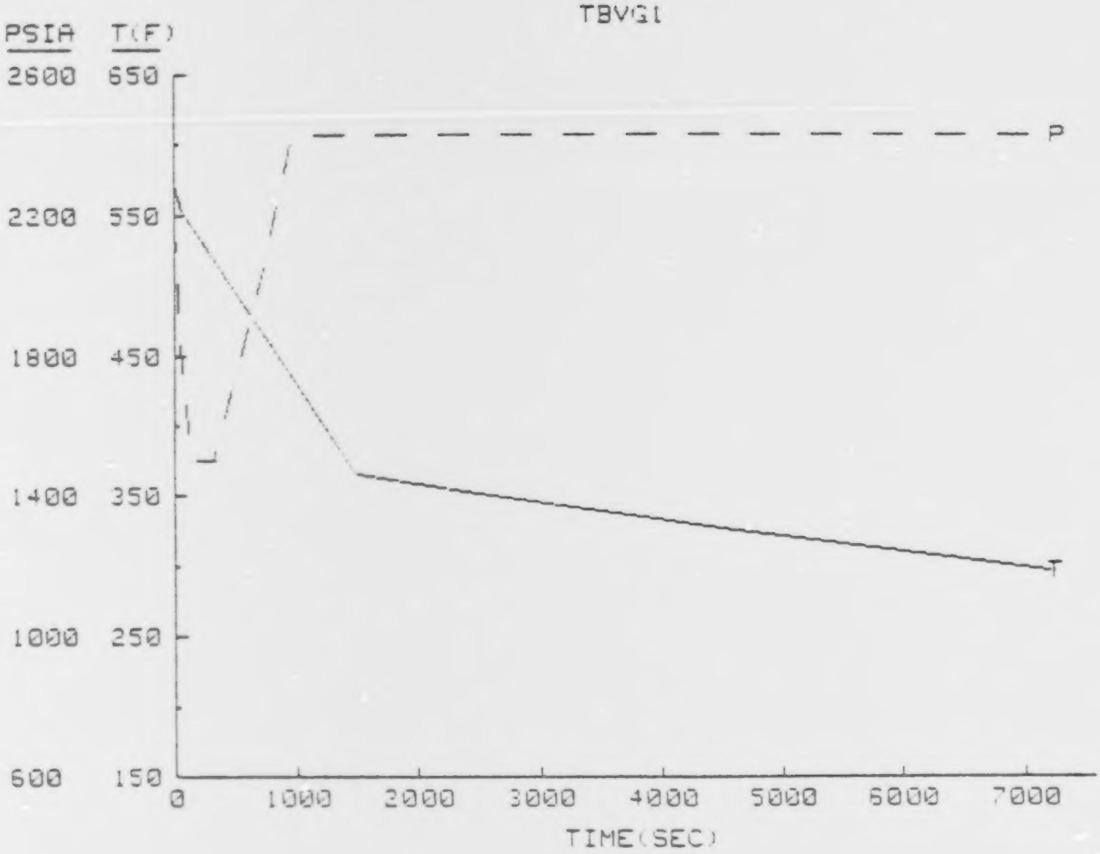

Figure C.9 Nominal pressures and temperatures for TBV group 1 sequences.

Table C.10 P, T, and h profiles for stagnated loop side of TBV group 1 sequences

TBVG1A .T		TBVG1A .P		TBVG1A .h	
Time(s)	T(F)	Time(s)	P(psia)	Time(s)	H(Btu/h) (ft**2F)
0.0	555.0	0.0	2180.0	0.0	2820.0
5.0	572.0	60.0	1740.0	183.0	2820.0
60.0	552.0	150.0	1500.0	233.0	210.0
1500.0	366.0	320.0	1500.0	7200.0	210.0
2500.0	346.5	710.0	2000.0		
3500.0	328.3	975.0	2430.0		
4500.0	311.2	7200.0	2430.0		
5500.0	295.1				
6500.0	280.0				
7200.0	270.0				

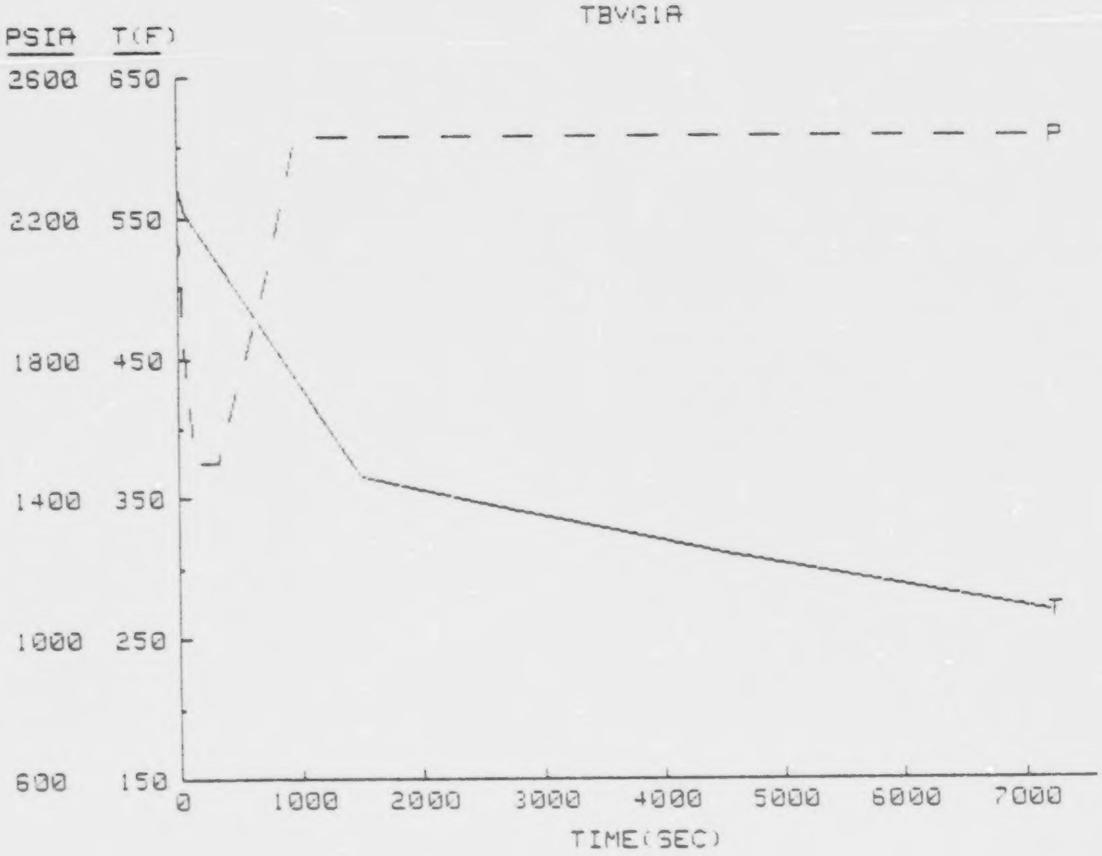

Figure C.10 Pressures and temperatures for stagnated loop side of TBV group 1 sequences.

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slightly different behavior because of PORV block valve closure. A final pressure of 2500 psi is attained at 1025 s when the safety relief valves are actuated. Note that the higher pressure would reduce HPI flow and cooldown because of that flow (see Tables C.9 and C.10 and Figures C.9 and C.10).

c. Heat Transfer Coefficient

The heat transfer coefficient is 2820 Btu/h ft^2 °F until the RCPs trip at 183 s. The coefficient then drops at 210 Btu/h ft^2 °F for the remainder of the transient (see Tables C.9 and C.10)

C.4.3 GROUP 2 -- Cases TBV3 and TBV5

a. Temperature

This group is taken to be identical to TBVG1 up through 22 min (1320 s) when one RCP/loop is restarted. SGB will supply heat to the primary to raise the downcomer temperature by an average of 20°F. Temperature will linearly decrease to 328°F at 2500 s as the 60,000-lb water inventory in SGA dries out. Steam flow restriction may increase the dryout period but would also result in higher final temperatures. After steam generator A dryout, the system will reheat at 0.0489°F/s to reach 547°F at 6980 s, and it would be limited at that point by automatic action of the TBVs in steam generator B (see Table C.11 and Figure C.11).

b. Pressure

The pressure curve of TBV group 1 (Figure C.9) was used to 1320 s. System shrinkage due to increased heat transfer caused pressure decrease on the order of 500 psi at the startup of the RCPs in LANL case 5C. Further drops were controlled by HPI injection. Similar behavior is assumed here, resulting in a pressure change from 2430 psi at 1320 s to 1930 psi at 1450 s that will continue at this value until SGA dryout at 2500 s. Then pressure is assumed to increase at 0.34 psi/s (observed value in LANL MSLB base case) because of system reheating until reaching PORV set point 2430 psi at 3950 s (see Table C.11 and Figure C.11).

c. Heat Transfer Coefficient

Heat transfer is the same as for TBV group 1 to 1320 s. With 1 RCP/loop flow established, the heat transfer coefficient becomes 1320 Btu/h $ft^{2o}F$ through the remainder of the transient (see Table C.11)

C.4.4 Group 3 -- Cases TBV4 and TBV6

a. Temperature

The massive MFW overfeeds in these scenarios will cause the transients to closely resemble LANL cases 5A and 5B to 1320 s (372°F) when 1 RCP/loop is restarted after feedwater isolation. A water inventory of 140,000 lb is predicted by LANL cases 5A and 5B at this point in the transient. The temperature would then decrease to 407°F at 3520 s when the 140,000-lb water inventory of SGA is finally exhausted. The system will then reheat at 0.0489°F/sec until reaching 547°F at 6380 s. Steam venting by the

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Table C.11 P, T, and h profiles for TBV group 2 sequences

TBVG2 .T		TBVG2 .P		TBVG2 .h	
Time(s)	T(F)	Time(s)	P(psia)	Time(s)	H(Btu/h)(ft**2F)
0.0	555.0	0.0	2180.0	0.0	2820.0
5.0	572.0	60.0	1740.0	183.0	2820.0
60.0	552.0	150.0	1500.0	233.0	210.0
1320.0	389.3	320.0	1500.0	1320.0	210.0
1321.0	409.3	710.0	2000.0	1321.0	1320.0
2500.0	328.0	975.0	2430.0	7200.0	1320.0
6980.0	547.0	1320.0	2430.0		
7200.0	547.0	1450.0	1930.0		
		2500.0	1930.0		
		3950.0	2430.0		
		7200.0	2430.0		

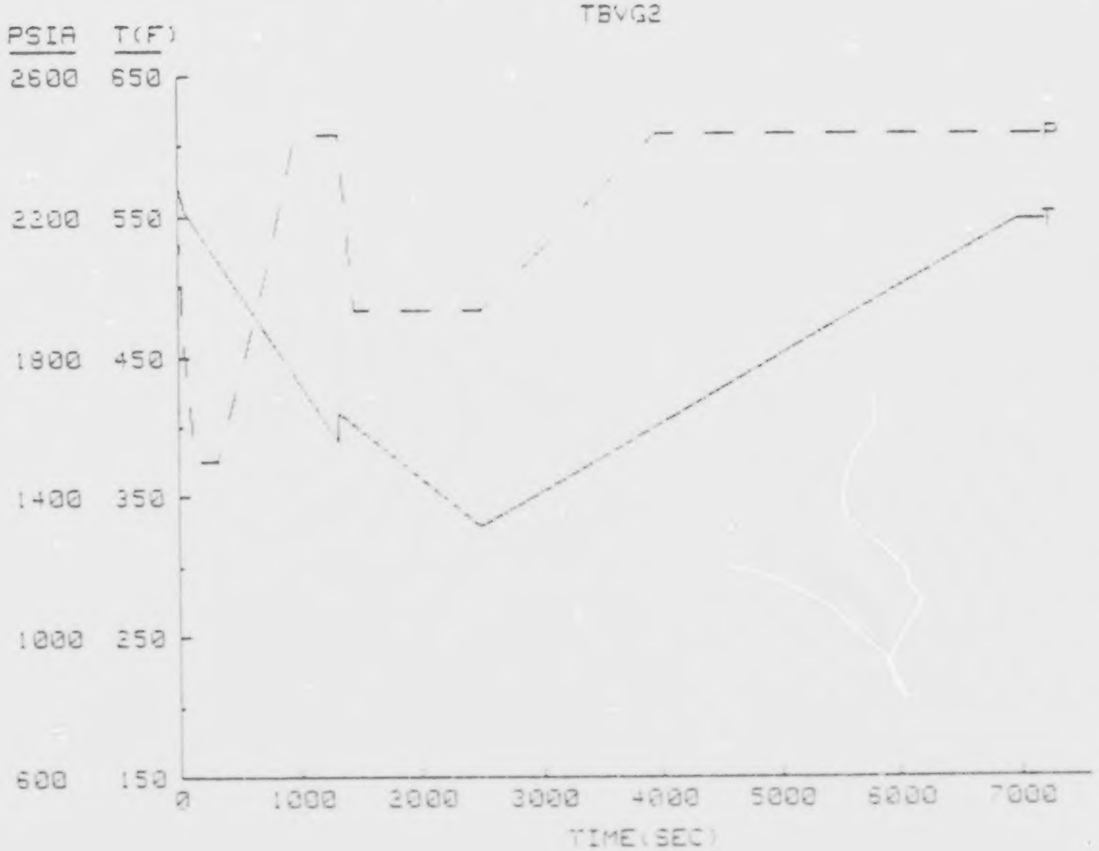

Figure C.11 Pressures and temperatures for TBV group 2 sequences.

SGB turbine bypass valves will keep the system at this temperature for the remainder of the transient (see Table C.12 and Figure C.12).

b. Pressure

The pressure will follow a similar trend as that of TBV group 2 except that repressurization after RCP startup will be delayed to 3520 s and final system pressure, 2430 psi, is regained at 4970 s (see Table C.12 and Figure C.12).

c. Heat Transfer Coefficient

This case is exactly the same as TBV group 2 (see Table C12).

C.4.5 Group 4 -- Case TBV7

a. Temperature

Because of the massive overfeed of cool water to both SGs, saturated heat transfer conditions are not expected and the case will closely resemble LANL case 5A, which is dominated by a sustained main and emergency feedwater overfeed through much of the sequence. The LANL-supplied values for LANL case 5A, including extrapolation to 7200 s, were applied to this case (see Table C.13 and Figure C.13).

If the HW/CB pump system does not sustain flow to the depressurized SGA, the open TBV would instead control the cooldown rate. However, higher temperatures ($t_{min} = 300^{o}F$ at 7200 s) would result.

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Table C.12 P, T, and h profiles for JBV group 3 sequences

TBVG3 .T		TBVG3 .P		TBVG3 .h	
Time(s)	T(F)	Time(s)	P(psia)	Time(s)	H(Btu/h) (ft**2F)
0.0	555.0	0.0	2180.0	0.0	2820.0
5.0	572.0	60.0	1740.0	183.0	2820.0
60.0	552.0	150.0	1500.0	233.0	210.0
158.0	531.0	320.0	1500.0	1320.0	210.0
316.0	470.0	710.0	2000.0	1321.0	1320.0
552.0	422.0	975.0	2430.0	7200.0	1320.0
1030.0	387.0	1320.0	2430.0		
1320.0	372.0	1450.0	1930.0		
1321.0	433.0	3520.0	1930.0		
3520.0	407.0	4970.0	2430.0		
6380.0	547.0	7200.0	2430.0		
7200.0	547.0				

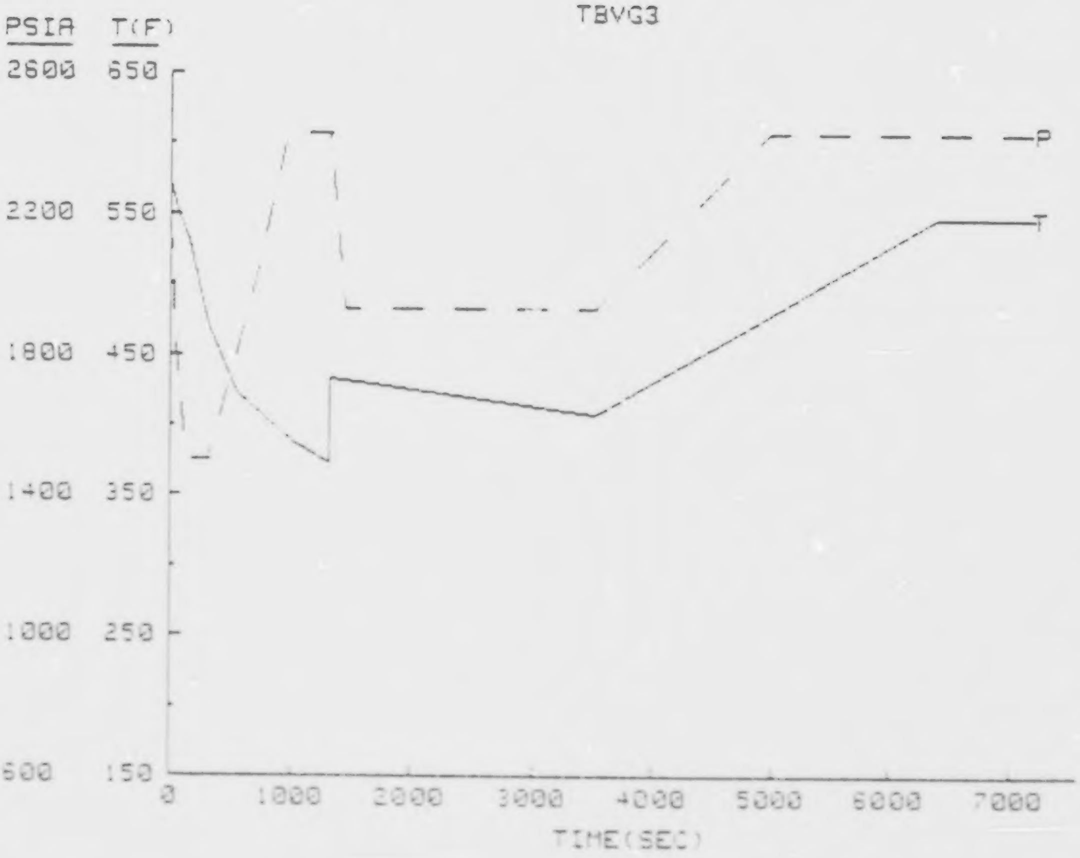

Figure C.12 Pressures and temperatures for TBV group 3 sequences.

Table C.13 P, T, and h profiles for TBV group 4 sequences

TBVG4 .T		TBVG4 .P		TBVG4 .h	
Time(s)	T(F)	Time(s)	P(psia)	Time(s)	H(Btu/h) (ft**2F)
0.0	555.0	0.0	2180.0	0.0	2820.0
5.0	572.0	60.0	1740.0	183.0	2820.0
60.0	552.0	150.0	1500.0	233.0	210.0
158.0	531.0	400.0	1500.0	7200.0	210.0
316.0	470.0	600.0	1750.0		
552.0	422.0	1036.0	2430.0		
1030.0	387.0	7200.0	2430.0		
1320.0	372.0				
1500.0	365.0				
7200.0	197.6				

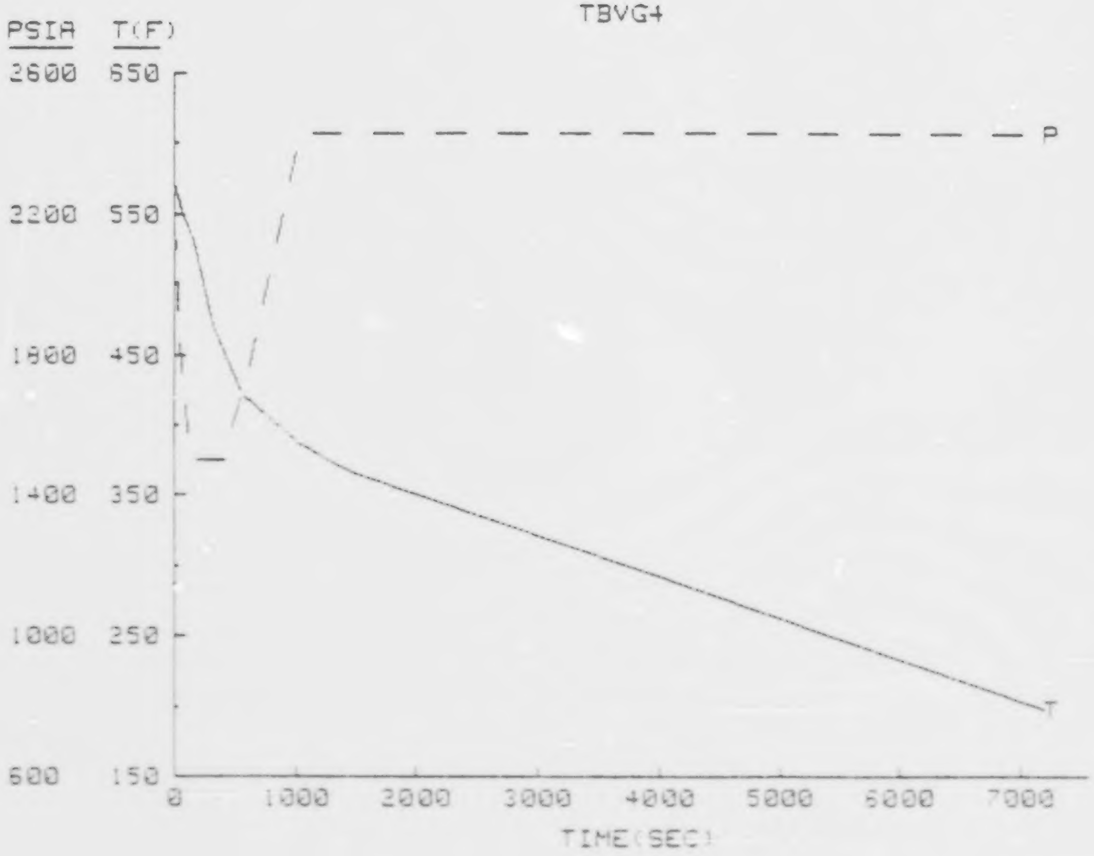

Figure C.13 Pressures and temperatures for TBV group 4 sequences.

b. Pressure

LANL case 5A as extrapolated was used for this case (see Table C.13 and Figure C.13).

c. Heat Transfer Coefficient

This case is identical to TBV group 1 (see Table C.13).

C.4.6 Group 5 -- Cases TBV8 and TBV16

a. Temperature

Group 5 follows LANL cases 6A, 6B, and 6C through the first 60 s. Afterwards, straight-line extrapolation is followed at the quasi-static temperature of 320°F at 1500 s. The cooldown model is followed to 265°F at 7200 s. Neither loop stagnates, so HPI mixing is not seen as a problem (see Table C.14 and Figure C.14)

b. Pressure

The pressure is assumed to drop to 1000 psia at 200 s where HPI limits further reductions. Pressure remains flat out to 500 s and then increases to the PORV set point (2430 psia) at 1175 s. Pressure remains at PORV limit for the remainder of the sequence (see Table C.14 and Figure C.14).

c. Heat Transfer Coefficient

No distinction is made between the one-loop TBV failures and the two-loop TBV failures. Therefore, this case is identical to TBV group 1 (see Table C.14).

Table C.14 P, T, and h profiles for TBV group 5 sequences

TBVG5 .T		TBVG5 .P		TBVG5 .h	
Time(s)	T(F)	Time(s)	P(psia)	Time(s)	H(Btu/h) (ft**2F)
0.0	555.0	0.0	2180.0	0.0	2820.0
5.0	572.0	200.0	1000.0	183.0	2820.0
60.0	552.0	500.0	1000.0	233.0	210.0
1500.0	320.0	1175.0	2430.0	7200.0	210.0
2500.0	309.4	7200.0	2430.0		
3500.0	299.3				
4500.0	289.5				
5500.0	280.1				
6500.0	271.0				
7200.0	265.0				

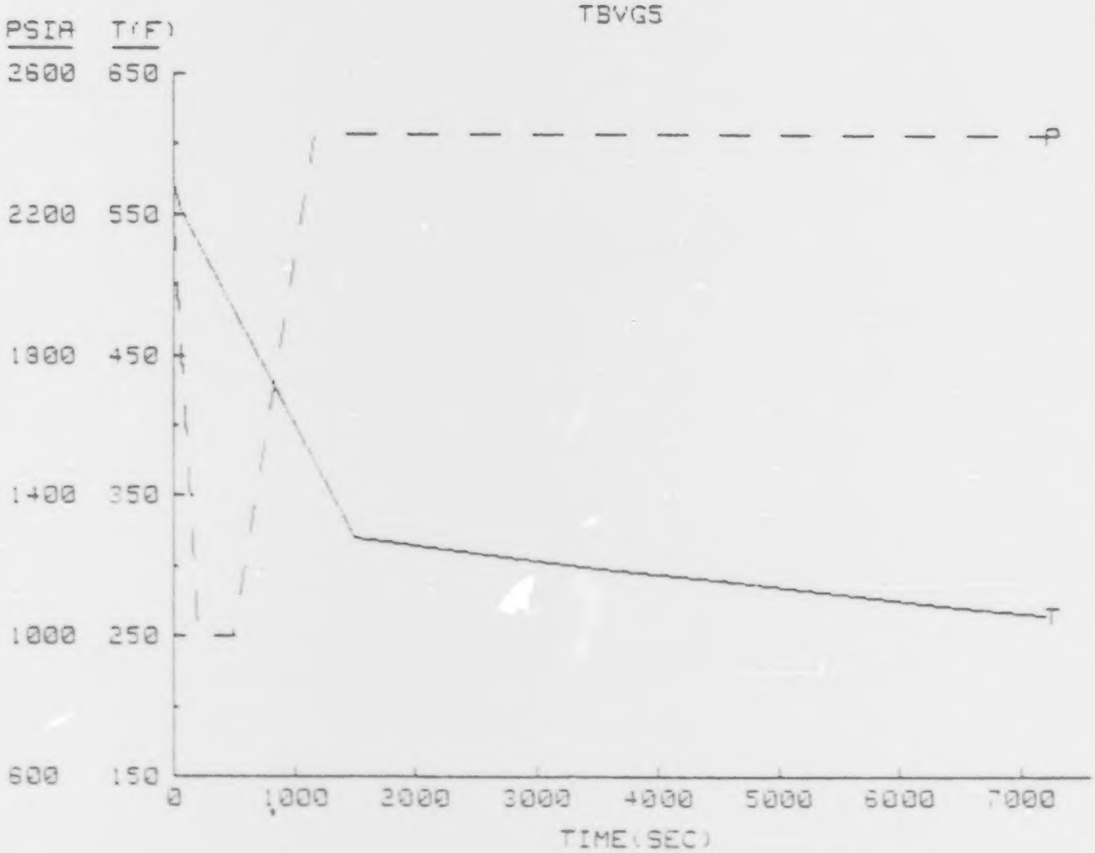

Figure C.14 Pressures and temperatures for TBV group 5 sequences.

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C.4.7 Group 6 -- Cases TBV10, TBV13, and TBV15

a. Temperature

These cases are similar to TBV group 1 to 975 s when the PORV actuates and sticks open. (TBV13 specifies failure of the PORV and a safety relief valve to close. However, the SRV would never be challenged and is therefore not assumed to fail.) HPI flow increases because of reduced primary pressure. The increase in flow will cause HPI flow to dominate over steam flow through the open TBV as the main heat removal mechanism late in the sequence. Based on loop, HPI, and vent valve flows from the LANL PORV-LOCA case 2A and TBV area and conditions described in Section C.4, a minimum temperature of 257°F was obtained at 7200 s using the cooldown model (see Table C.15 and Figure C.15).

b. Pressure

This group is similar to TBV group 1 to 975 s when the PORV is first actuated. The pressurizer is liquid-solid at this point. Failure of the PORV to close results in water flow through the valve and a drop in pressure, which promotes more flow. The pressure decreases and the flow increases until an equilibrium is struck between PORV flow resistance and HPI head capacity characteristics. This equilibrium point is at a system pressure of 1670 psia, as obtained from LANL case 2A. No voiding is expected as a result of this depressurization. The system reaches this final pressure quickly (assume 25 s) and remains there for the rest of the transient (see Table C.15 and Figure C.15).

Table C.15 P, T, and h profiles for TBV group 6 sequences

TBVG6 .T		TBVG6 .P		TBVG6 .h	
Time(s)	T(F)	Time(s)	P(psia)	Time(s)	H(Btu/h) (ft**2F)
0.0	555.0	0.0	2180.0	0.0	2820.0
5.0	572.0	60.0	1740.0	183.0	2820.0
60.0	552.0	150.0	1500.0	233.0	210.0
975.0	433.0	320.0	1500.0	7200.0	210.0
1500.0	373.0	710.0	2000.0		
2500.0	335.1	975.0	2430.0		
3500.0	308.7	1000.0	1670.0		
4500.0	289.4	7200.0	1670.0		
5500.0	274.8				
6500.0	263.6				
7200.0	257.0				

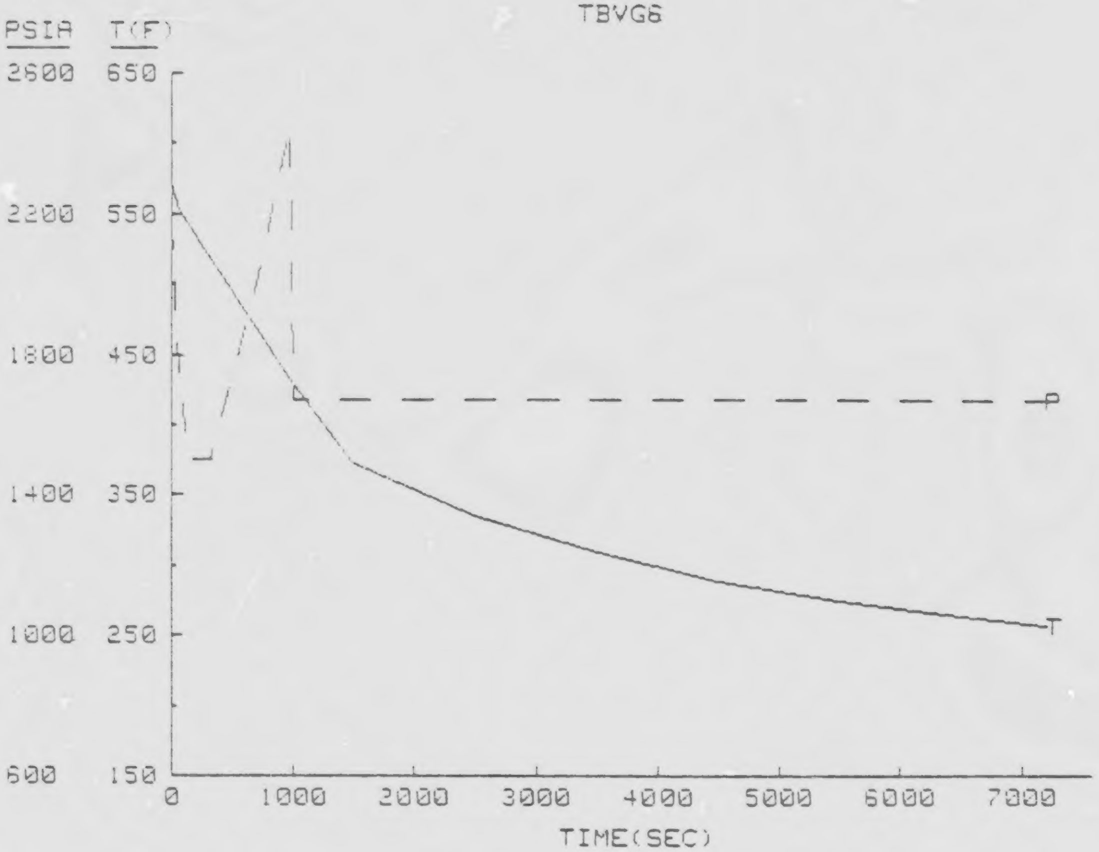

Figure C.15 Pressures and temperatures for TBV group 6 sequences.

c. Heat Transfer Coefficient

This group is identical to file TBV group 1 (see Table C.15).

C.4.8 Group 7 -- Case TBV12

a. Temperature

The combination of a single TBV failure and a single steam safety relief valve on loop A makes group 7 similar to a TBV bank failure, as in LANL cases 5A, 5B, and 5C. However, the proper operation of the MFW system would preclude the severe overfeeding late in LANL cases 5A, 5B, and 5C. Therefore, cooldown rates calculated from the expanded cooldown model were used to predict the system response. LANL case 5A was used to 60 s (552°F). Extrapolation by the cooldown model yielded a minimum temperature of 272°F at 7200 s. This temperature response is very similar to that of TBV group 5, which features the same total break area apportioned between both loops. The main differences between the two groups are due to different loop flow and vent valve flow rates (see Table C.16 and Figure C.16).

b. Pressure

The group is taken to be identical to TBV group 1 (see Table C.16 and Figure C.16).

c. Heat Transfer Coefficient

The group is taken to be identical to TBV group 1 (see Table C.16).

Table C.16 P, T, and h profiles for TBV group 7 sequences

TBVG7N.T		TBVG7N.P		TBVG7N.h	
Time(s)	T(F)	Time(s)	P(psia)	Time(s)	H(Btu/h) (ft**2F)
0.0	555.0	0.0	2180.0	0.0	2820.0
5.0	572.0	60.0	1740.0	183.0	2820.0
60.0	552.0	150.0	1500.0	233.0	210.0
400.0	392.9	320.0	1500.0	7200.0	210.0
500.0	386.2	710.0	2000.0		
600.0	380.0	975.0	2430.0		
700.0	376.5	7200.0	2430.0		
800.0	371.4				
900.0	368.3				
1000.0	364.8				
1500.0	353.9				
2500.0	330.0				
3500.0	311.5				
4500.0	297.2				
5500.0	286.0				
6500.0	277.2				
7200.0	272.2				

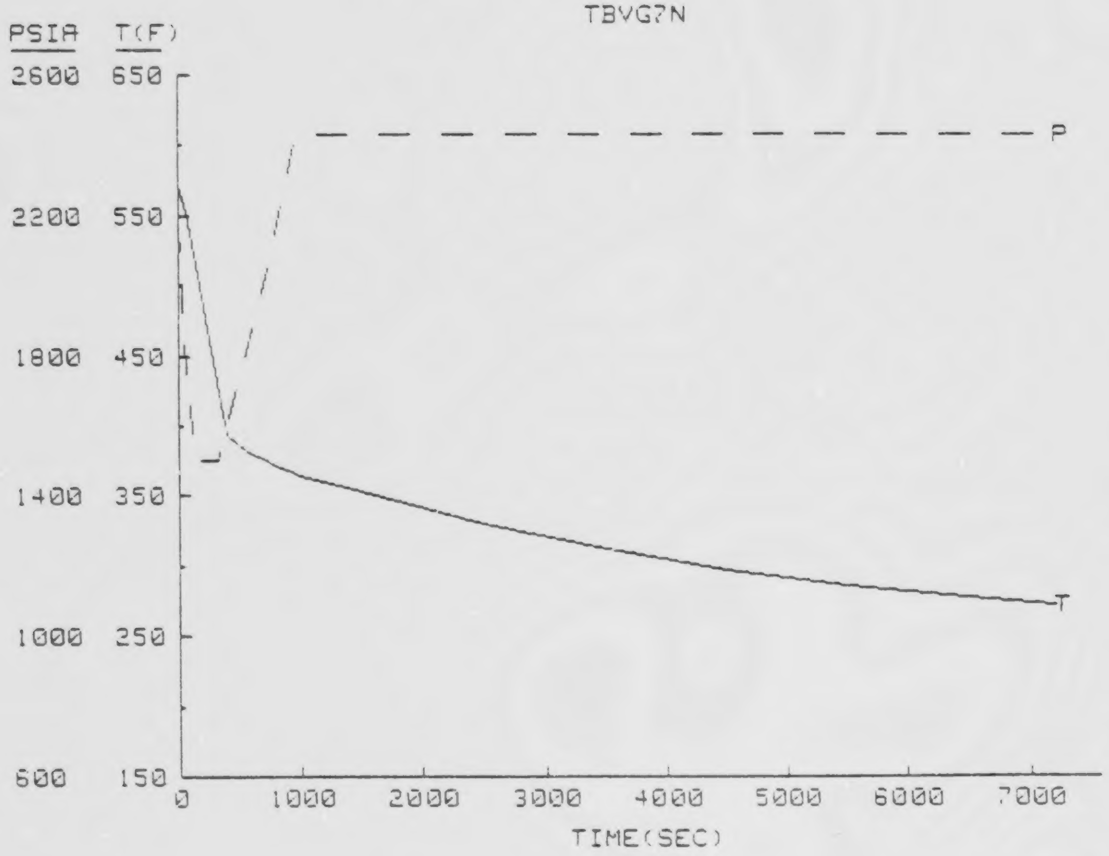

Figure C.16 Pressures and temperatures for TBV group 7 sequences.

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C.4.9 Group 8 -- Case TBV11

a. Temperature

This group differs from TBV group 7 in that the feedwater is isolated at 20 min and 1 RCP/loop is restarted at 22 min (1320 s). Therefore, the case uses the extrapolation for TBV group 7 out to 1320 s. The restart of the RCPs causes mixing of hot water from SGB with the other water in the system to raise the downcomer temperature to 433°F. Flow choking limits the steam loss rate in SGA to about 100 lb/s. SGA dryout is accomplished at 1820 s, at which time system temperature is 423°F, the level necessary to maintain the steam loss rate in SGA. After SGA dryout, the system reheats at 0.0489°F/s to reach 547°F at 4360 s, where it remains for the rest of the sequence (see Table C.17 and Figure C.17).

b. Pressure

This case is taken to be identical to TBV group 3 to 1450 s. The pressure to this point has fallen from 2180 to 1500 psia at 150 s, risen to 2430 psia at 975 s, and dropped to 1930 psia at 1450 s because of mixing of hot and cool liquids in the primary after RCP startup. The pressure is assumed constant to 1820 s and then increases at 0.3446 psi/s to 2430 psia at 3270 s, where it remains for the rest of the sequence (see Table C.17 and Figure C.17).

c. Heat Transfer Coefficient

This group is taken to be identical to TBV group 2 (see Table C.17).

Table C.17 P, T, and h profiles for TBV group 8 sequences

TBVG8 .T		TBVG8 .P		TBVG8 .h	
Time(s)	T(F)	Time(s)	P(psia)	Time(s)	H(Btu/h) (ft**2F)
0.0	555.0	0.0	2180.0	0.0	2820.0
5.0	572.0	60.0	1740.0	183.0	2820.0
60.0	552.0	150.0	1500.0	233.0	210.0
1320.0	348.7	320.0	1500.0	1320.0	210.0
1321.0	433.0	710.0	2000.0	1321.0	1320.0
1820.0	423.0	975.0	2430.0	7200.0	1320.0
4360.0	547.0	1320.0	2430.0		
7200.0	547.0	1450.0	1930.0		
		1820.0	1930.0		
		3270.0	2430.0		
		7200.0	2430.0		

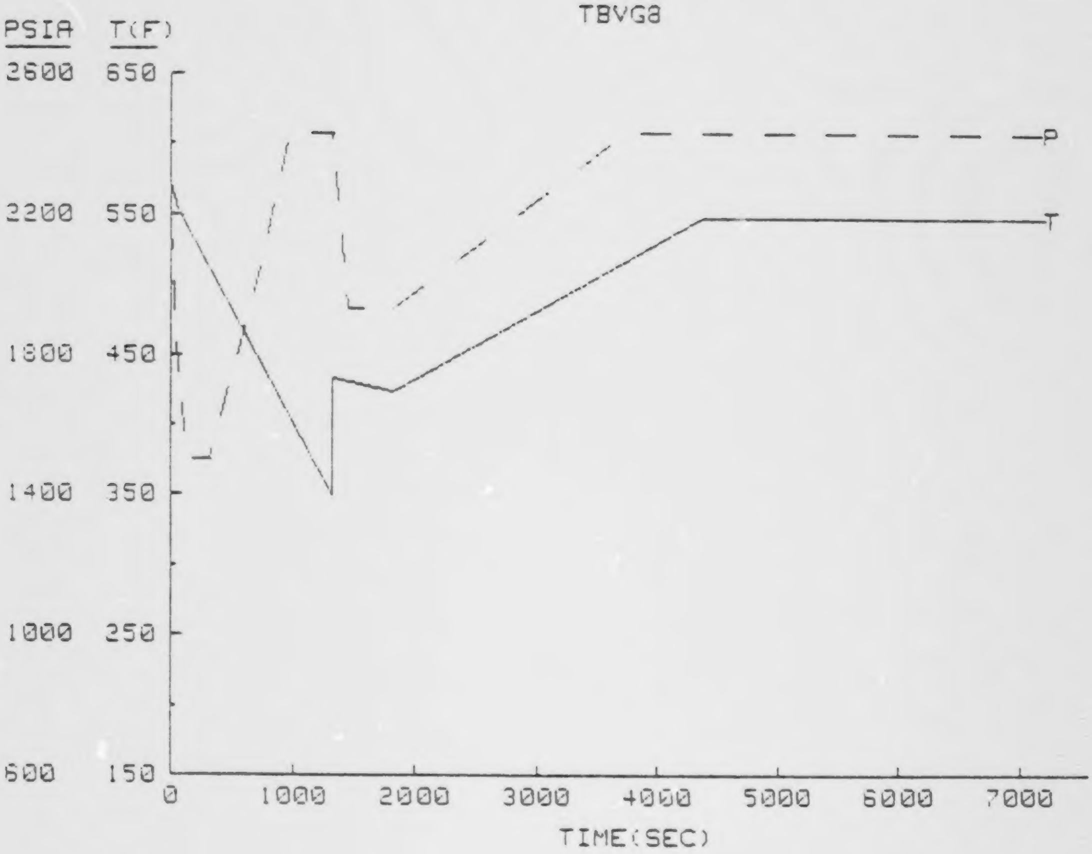

Figure C.17 Pressures and temperatures for TBV group 8 sequences.

C.4.10 Group 9 -- Case TBV17

a. Temperature

With the MFW overfeed and subsequent flooding of SGA with flow from the HW/CB pumps via the startup flow control valves, the case should closely resemble LANL case 5B. After SGA is completely full, feedwater will flow from the feedwater ring to the SG outlet with a small but undetermined amount of heat being absorbed from the primary. For this analysis, it is assumed that this follows a single loop TBV failure response identical to LANL case 5A, as extrapolated (see Table C.18 and Figure C.18).

b. Pressure

This group is assumed to be identical to TBV group 5 because of early effects of system cooldown (see Table C.18 and Figure C.18).

c. Heat Transfer Coefficient

This group is taken to be identical to TBV group 1 (see Table C.18).

C.4.11 Group 10 -- Cases TBV18 and TBV19

a. Temperature

This group includes the equivalent of single TBV failures in each loop and a PORV-sized LOCA. The case is similar to TBV group 6 except that both loops are affected. LANL case 5A was used to 60 s (552°F), then the cooldown model was used to extrapolate to a minimum temperature of 236.7°F at 7200 s (see Table C.19 and Figure C.19).

Table C.18 P, T, and h profiles for TBV group 9 sequences

TBVG9 .T		TBVG9 .P		TBVG9 .h	
Time(s)	T(F)	Time(s)	P(psia)	Time(s)	H(Btu/h) (ft**2F)
0.0	555.0	0.0	2180.0	0.0	2820.0
5.0	572.0	200.0	1000.0	183.0	2820.0
60.0	552.0	500.0	1000.0	233.0	210.0
158.0	531.0	1175.0	2430.0	7200.0	210.0
316.0	470.0	7200.0	2430.0		
552.0	422.0				
1030.0	387.0				
1320.0	372.0				
1500.0	365.0				
7200.0	197.6				

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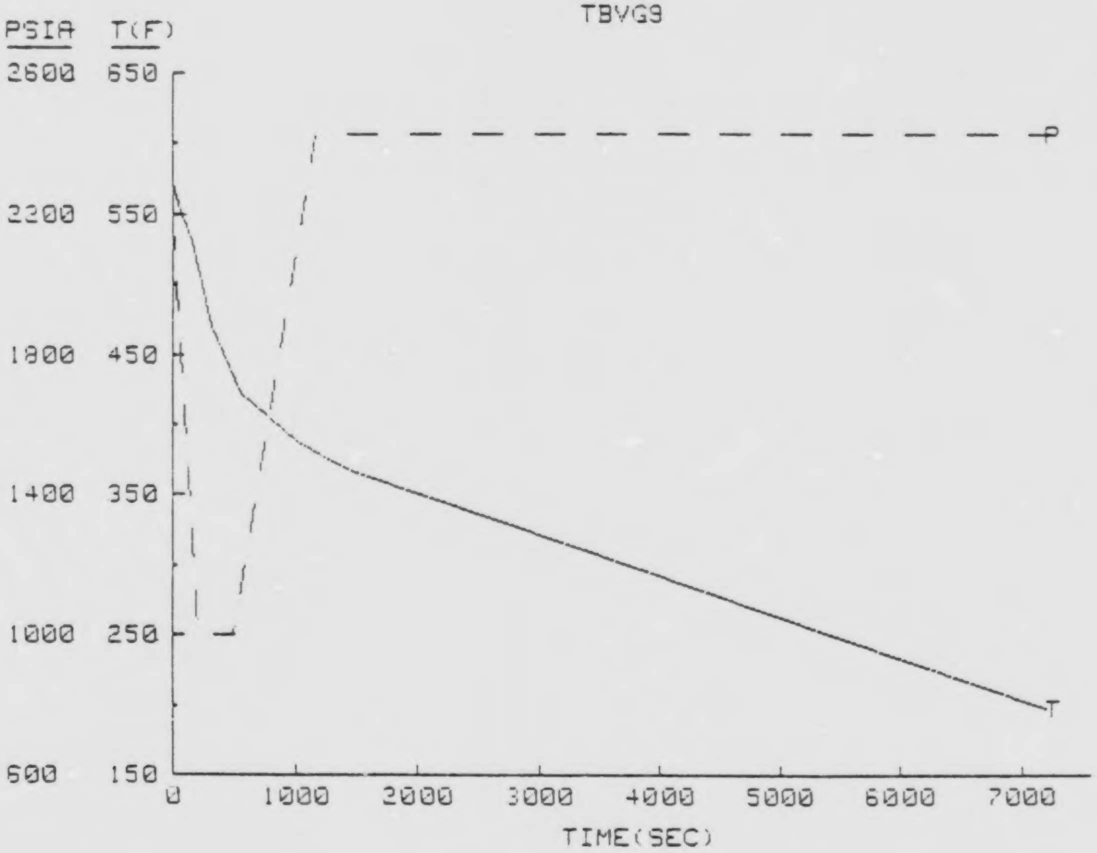

Figure C.18 Pressures and temperatures for TBV group 9 sequences.

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Table C.19 P, T, and h profiles for TBV group 10 sequences

TBVG10.T		TBVG10.P		TBVG10.h	
Time(s)	T(F)	Time(s)	P(psia)	Time(s)	H(Btu/h) (ft**2F)
0.0	555.0	0.0	2180.0	0.0	2820.0
5.0	572.0	200.0	1000.0	183.0	2820.0
60.0	552.0	500.0	1000.0	233.0	210.0
1500.0	331.1	1175.0	2430.0	7200.0	210.0
2500.0	296.6	1200.0	1670.0		
3500.0	275.2	7200.0	1670.0		
4500.0	260.4				
5500.0	249.5				
6500.0	241.3				
7200.0	236.7				

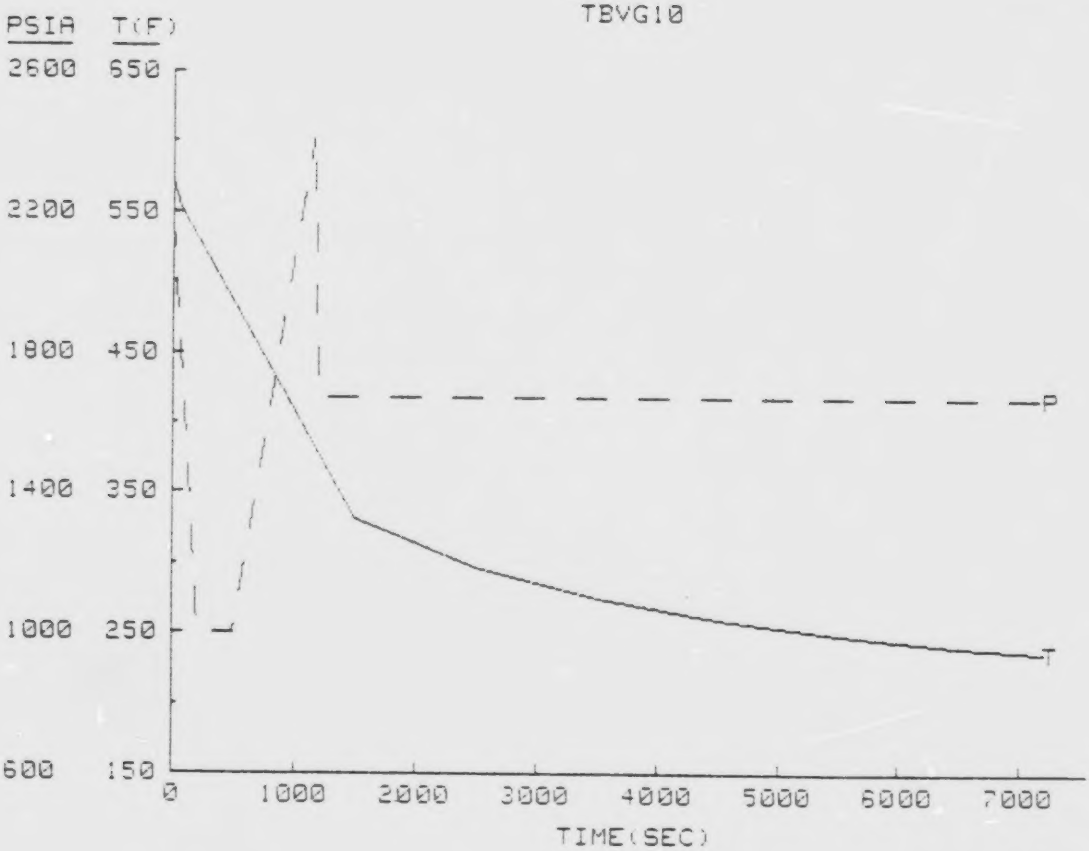

Figure C.19 Pressures and temperatures for TBV group 10 sequences.

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b. Pressure

This group resembles TBV group 5 out to 1175 s. The pressure has dropped from 2180 psia to 1000 psia at 200 s under the influence of overcooling both loops. At 500 s, pressure begins to increase from 1000 psia to 2430 psia at 1175 s. PORV sticks open at this point, and pressure drops to 1670 psia at 1200 s as HPI flow increases to sustain pressure at this level for the remainder of the sequence (see Table C.19 and Figure C.19).

c. Heat Transfer Coefficient

This group is identical to TBV group 1 (see Table C.19).

C.5 Turbine Bypass Valve Failures in Hot Standby

Overcooling events at hot standby can yield severe cooldown rates because of the absence of significant decay heat input to the system. INEL performed a calculation of a sequence where all four TBVs fail open (because of a control-system fault) and are isolated 10 min later. ORNL specified cases where one TBV and two TBVs in one loop fail open without subsequent isolation of the TBV(s) or throttling of HPI. To investigate the sensitivity of the sequences to decay heat level, decay heat rates corresponding to 12 h and 1 wk respectively since shutdown from full power. Table C.20 describes the sequences. The cooldown model predicted very rapid cooldown rates, largely because of poor accounting of pipe and vessel wall-to-water heat transfer. The error is less pronounced for transients from full power where decay heat-induced natural circulation and vent valve flows swamp any metal-to-water heating effects.

C.5.1 Basis

INEL -- 4TBV failure at hot standby.

C.5.2 Departures from Basis

a. One- and two-TBV failures were evaluated.

b. TBVs in base case were isolated at 10 min.
Study cases have no TBV isolation.

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Table C.20 Hot standby TBV failure cases

Conditions	Case ZP 1TBV (12 h)	 ZP 2TBV (12 h)	 ZW 1TBV (1 wk)	 ZW 2TBV (1 wk)
Initial Conditions				
Hot standby	x	x	x	x
Decay heat levels				
a. 12 hr = 15.75 MW	x	x		
b. 1 wk = 7.5 MW			x	x
Secondary Pressure Control				
1 TBV opens SGA	x		x	
2 TBVs open SGA		x		x
SGB TBVs operate as designed	x	x	x	x
SSRVs operate as designed	x	x	x	x
Feedwater Supply				
Feedwater controlled to appropriate level*	x	x	x	x
Primary Pressure Control				
PORV operates as designed	x	x	x	x
Operator Actions				
RCPs tripped at HPI + 30 s	x	x	x	x

*Assumes post-RCP trip SG level maintained by whatever feedwater source available

C.5.3 Applicable Data from Other Cases

None.

C.5.4 Extrapolation Assumptions, Procedures, and Results

a. Temperature

The cooldown model was employed to evaluate single- and double-TBV failures at hot standby. Total loop flows of 40,000 lb/s were used prior to RCP trip, and a total natural circulation rate of 700 lb/s was used in later stages of the transients. These values were obtained from the INEL base case calculation. Two power levels were examined. The first level used a constant residual decay heat of 15.7 MW, corresponding to about 12-14 h after trip from full-power operation. The second level used a value of 8.5 MW, corresponding to a shutdown period of 1 wk. A small vent valve flow of 45 lb/s was taken from the LANL PORV LOCA case. Initial primary temperatures of 548°F and feedwater temperatures of 90°F were assumed for both the MFW and EFW systems.

For one TBV failure at 12 h after shutdown, downcomer temperature decreases to 498°F at 250 s when HPI comes on, to 328°F at 480 s when SG fills to 20 ft of water after RCP trip is completed, and then to 303°F at 1000 s when the primary system goes water-solid and HPI flow decreases. The downcomer temperature rises to 307°F at 1500 s because of the HPI flow decrease relative to the warmer loop flows, but then it resumes decline, reaching 300°F at 2000 s and 234°F at 7200 s (see Table C.21 and Figure C.20).

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Table C.21 P, T, and h profiles for 1 TBV failure at hot standby (12 h after shutdown)

ZP1TBV.T		ZP1TBV.P		ZP1TBV.h	
Time(s)	T(F)	Time(s)	P(psia)	Time(s)	H(Btu/h) (ft**2F)
0.0	548.0	0.0	2180.0	0.0	2820.0
150.0	513.1	250.0	1500.0	280.0	2820.0
250.0	497.6	700.0	1500.0	330.0	300.0
280.0	490.3	1000.0	2430.0	2000.0	300.0
480.0	328.0	7200.0	2430.0	7200.0	147.0
1000.0	302.8				
1500.0	306.8				
2000.0	299.7				
2500.0	285.6				
3000.0	276.9				
4000.0	262.6				
5000.0	251.5				
6000.0	242.6				
7200.0	234.3				

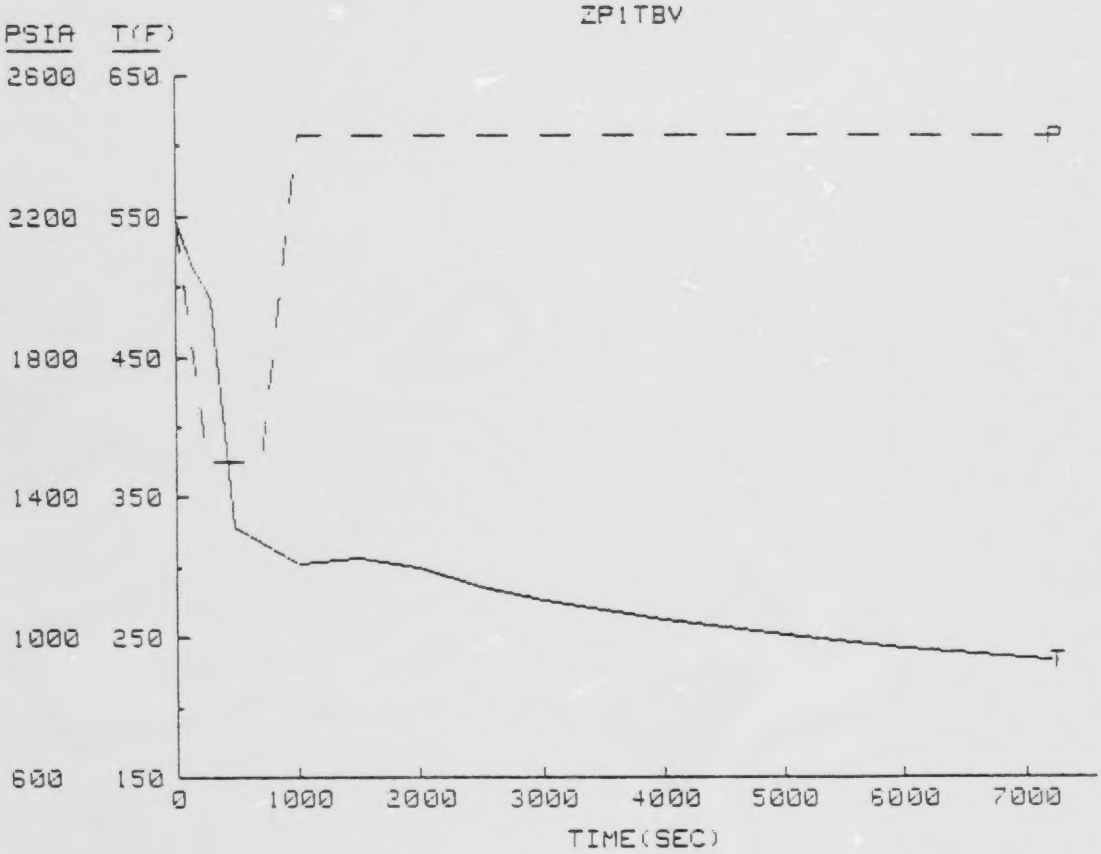

Figure C.20 Pressures and temperatures for 1 TBV failure at hot standby (12 hrs. after shutdown).

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For one TBV failed open 1 wk after shutdown, temperature decreases to 501°F at 200 s when HPI comes on, to 327°F at 460 s when the SGs fill to 20 ft of water after RCP trip, and then to 298°F at 1000 s when the primary goes water-solid and HPI flow decreases. The downcomer temperature rises to 300°F at 1500 s because of reduced HPI flow (temperature dilution), but then it resumes decline as the system cooldown overtakes the HPI mixing as the dominant mechanism. The downcomer temperature drops to 286.4°F at 2000 s and 207.2°F at 7200 s. The difference in decay heat input is most noticeable late in the sequence (see Table C.22 and Figure C.21).

With two TBVs fail open 12 h after shutdown, temperature declines to 490°F at 120 s when the HPI comes on, to 303°F at 350 s when SGs fill to 20 ft of water after RCP trip is complete, and to 277°F at 900 s when the primary system goes water-solid. The downcomer temperature is assumed to stay essentially constant to 1500 s in response to reduced HPI flow. Then temperature decline resumes, and a final value of 209°F is reached at 7200 s (see Table C.23 and Figure C.22).

For two TBVs failed open 1 wk after shutdown, temperature declines to 486.8°F at 120 s when SG fill is completed after RCP trip and to 269°F at 1000 s when the primary goes water-solid and HPI flows decrease. Temperature remains relatively constant to 1500 s because of the reduction in HPI flow. Then the temperature decline resumes, with a final value of 185°F reached at 7200 s. Again, the difference in decay input is only discernable late in the sequence (see Table C.24 and Figure C.23).

Table C.22 P, T, and h profiles for 1 TBV failure at hot standby (1 wk after shutdown)

ZW1TBV.T		ZW1TBV.P		ZW1TBV.h	
Time(s)	T(F)	Time(s)	P(psia)	Time(s)	H(Btu/h) (ft**2F)
0.0	548.0	0.0	2180.0	0.0	2820.0
50.0	530.1	230.0	1500.0	260.0	2820.0
100.0	518.4	700.0	1500.0	310.0	300.0
200.0	501.2	1000.0	2430.0	2000.0	300.0
230.0	496.7	7200.0	2430.0	7200.0	147.0
260.0	488.4				
360.0	346.6				
460.0	327.6				
600.0	319.3				
800.0	308.5				
1000.0	298.7				
1500.0	300.0				
2000.0	286.4				
2500.0	274.4				
3000.0	263.7				
3500.0	254.2				
4000.0	245.7				
4500.0	238.0				
5000.0	231.1				
5500.0	224.8				
6000.0	219.0				
6500.0	213.8				
7000.0	209.0				
7200.0	207.2				

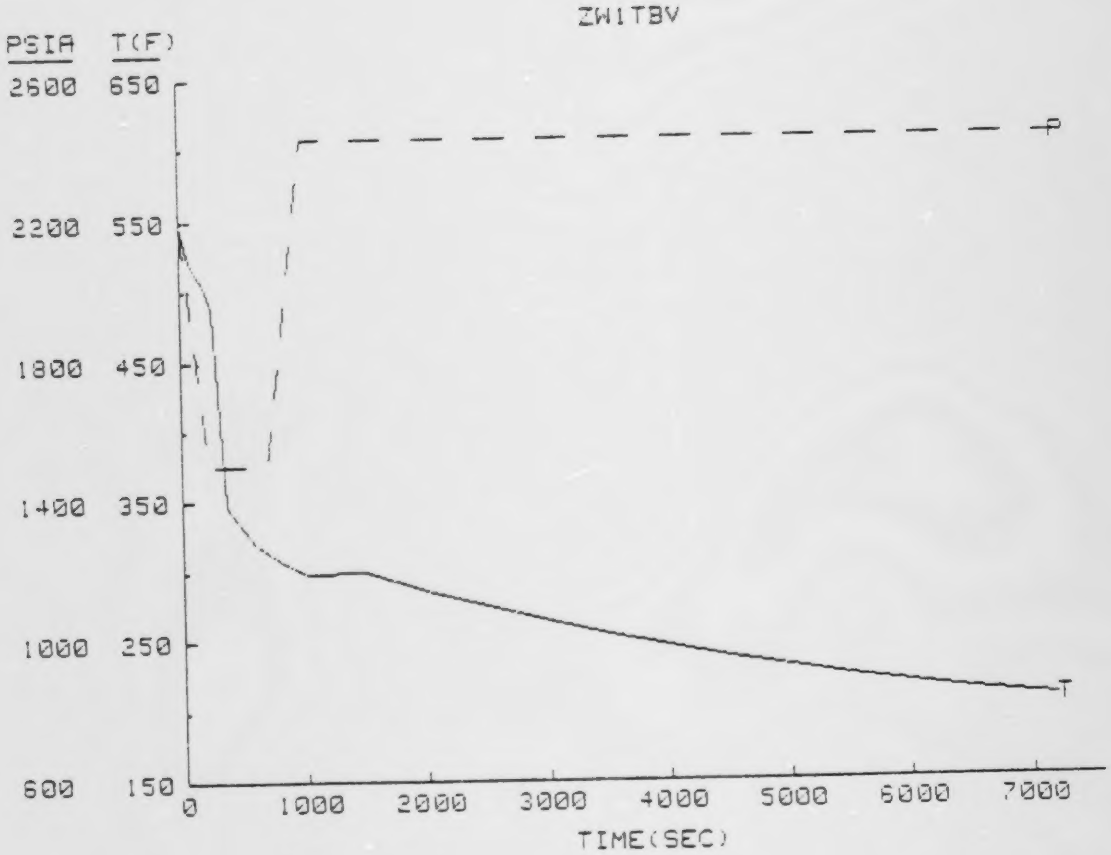

Figure C.21 Pressures and temperatures for 1 TBV failure at hot standby (1 wk after shutdown).

Table C.23 P, T, and h profiles for 2-TBV failure at hot standby (12 h after shutdown)

ZP2TBV.T		ZP2TBV.P		ZP2TBV.h	
Time(s)	T(F)	Time(s)	P(psia)	Time(s)	H(Btu/h) (ft**2F)
0.0	548.0	0.0	2180.0	0.0	2820.0
120.0	490.7	120.0	1500.0	150.0	2820.0
150.0	478.5	600.0	1500.0	200.0	300.0
350.0	303.1	900.0	2430.0	2000.0	300.0
900.0	276.7	7200.0	2430.0	7200.0	147.0
1500.0	275.8				
2000.0	264.5				
2500.0	255.0				
3000.0	246.7				
4000.0	233.5				
5000.0	223.4				
6000.0	215.8				
7200.0	208.7				

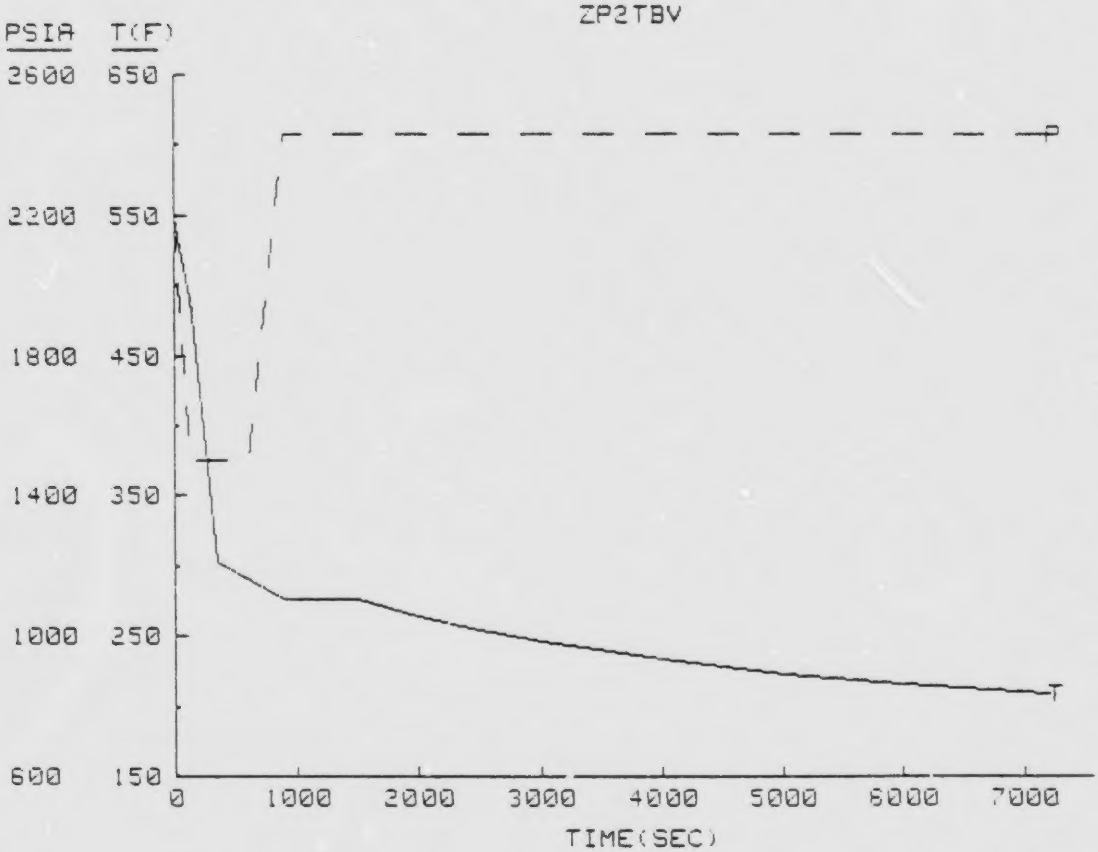

Figure C.22 Pressures and temperatures for 2 TBV failure at hot standby (12 hrs. after shutdown).

Table C.24 P, T, and h profiles for 2- TBV failure at hot standby (1 wk after shutdown)

ZW2TBV.T		ZW2TBV.P		ZW2TBV.h	
Time(s)	T(F)	Time(s)	P(psia)	Time(s)	H(Btu/h) (ft**2F)
0.0	548.0	0.0	2180.0	0.0	2820.0
60.0	507.1	120.0	1500.0	150.0	2820.0
120.0	486.8	700.0	1500.0	200.0	300.0
150.0	474.5	1000.0	2430.0	2000.0	300.0
200.0	328.9	7200.0	2430.0	7200.0	147.0
250.0	319.5				
300.0	310.7				
350.0	302.6				
500.0	293.9				
700.0	283.1				
850.0	275.9				
1000.0	269.2				
1500.0	268.9				
2000.0	256.0				
2500.0	244.7				
3000.0	234.8				
3500.0	226.2				
4000.0	218.5				
4500.0	211.6				
5000.0	205.5				
5500.0	200.0				
6000.0	195.1				
6500.0	190.7				
7000.0	186.6				
7200.0	185.0				

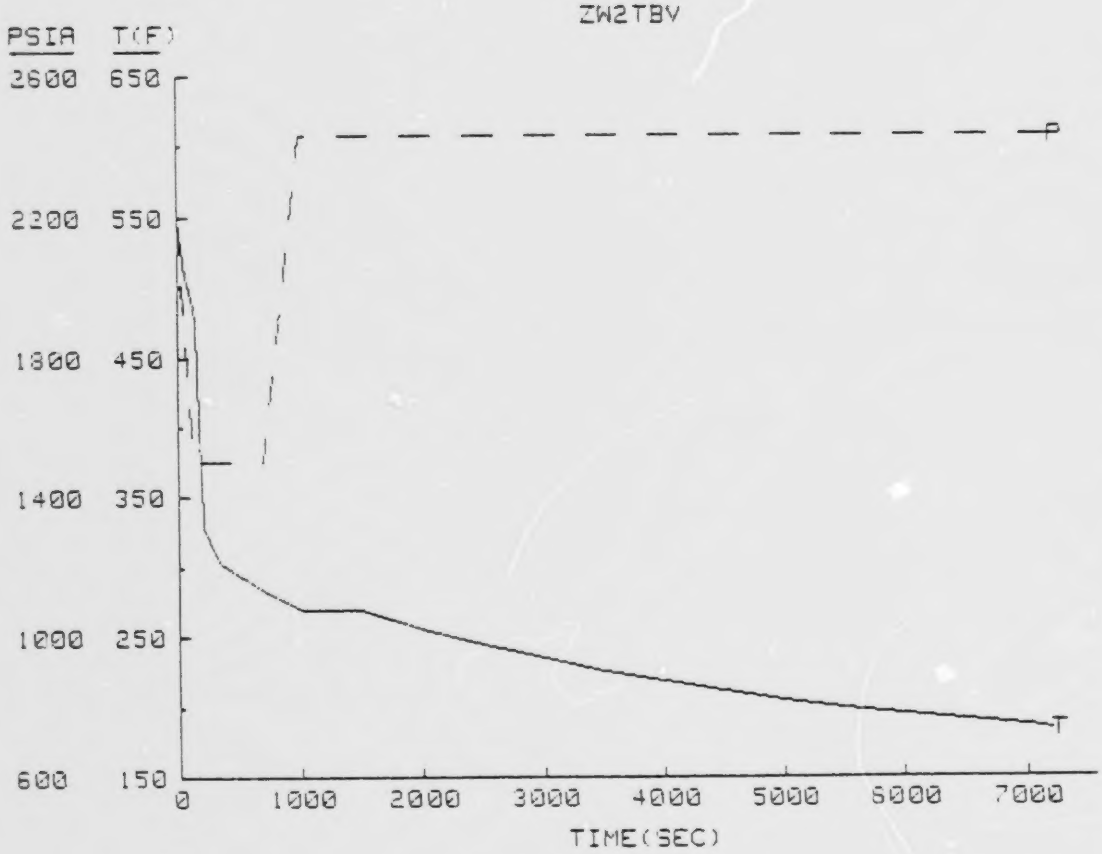

Figure C.23 Pressures and temperatures for 2 TBV failure at hot standby (1 wk after shutdown).

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Minimum temperatures lower than the base case were obtained for these study cases because the TBVs were not isolated at 600 s, as in the INEL base case.

b. Pressure

The time to reach HPI set point was assumed to be a function of average coolant temperature. However, the action of pressurizer heaters and makeup/letdown systems greatly complicates the situation. Starting from hot standby conditions, the pressurizer should empty when the average temperature drops below 500°F. It is assumed that the pressure is near the HPI set point when the pressurizer empties. With one TBV open, the system depressurizes (cools) to the HPI set point (1500 psia) at 230 s (1-wk shutdown) or 250 s (12-h shutdown). HPI is assumed to maintain pressure at this level until 700 s, whereon pressure begins to rise to the PORV setpoint, 2430psi, at 1000 s where it remains throughout the sequence (see Tables C.21 and C.22 and Figures C.20 and C.21).

With two TBVs open, the system reaches HPI set point at 120 s (similar to base case). Instead of decreasing further, it is assumed that pressure remains at 1500 psia until 600 s (12-h shutdown) or 700 s (1-wk shutdown), when it begins to rise to its final value of 2430 psia at 900 s (12-h shutdown) or 1000 s (1-wk shutdown). (See Tables C.23 and C.24 and Figures C.22 and C.23.)

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c. Heat Transfer Coefficient

For both cases, an initial value of 2820 (Btu/h/ft^{2o}F) was applied from the LANL calculations. After RCP trip, INEL gives values of 300 (Btu/h/ft^{2o}F) to 2000 s, which then decrease to 147 (Btu/h/ft^{2o}F) at 7200 s. This type of gradual decrease was expected but not observed in any other sequences. No correction for free convection effects is included in these values (see Tables C.21 through C.24).

C.6 PORV-Sized LOCA Cases

C.6.1 ORNL-Defined PORV-Sized LOCAs

ORNL defined nine PORV-sized LOCA cases. These were sorted into five groups based on case similarities. The characteristics of the nine cases and their respective groupings are displayed in Table C.25. The transients are evaluated by groups below.

C.6.1.1 Bases

LANL PORV LOCA case 2A

LANL Rancho Seco-type transient

LANL TBV failure case 5B

C.6.1.2 Departures from Bases

a. All cases have a PORV-sized break, but not necessarily a failed PORV. Location of the break in the pressurizer surge line (for example), rather than at the top of the pressurizer, would change system response since the system may not go water-solid. However, a stuck-open PORV or an equivalent break at the top of the pressurizer would result in larger cumulative HPI flows and hence greater cooldown than a similar-size break elsewhere.

b. Feedwater perturbations such as normal runback, SGA overfeeds, and loss of MFW with EFW failure are examined. Normal runback is not covered in any of the bases.

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Table C.25 PORV-sized LOCA cases and groupings

Condition[*]	Case SBLOCA- 1	2	3	4	5	6	7	8	9
Secondary Pressure Controls									
Operates as designed	x	x	x	x	x		x	x	x
2 TBVs fail open loop A						x			
Feedwater									
MFW runback as designed	x					x	x		
SGA runback failure		x		x				x	
LOMFW at time = 0.0			x		x				x
EFW operates as designed	x	x	x			x	x	x	x
EFW in manual				x	x				
Primary Pressure Controls									
PORV block valve open	x	x	x	x	x	x	x	x	x
PORV operates as designed[**,***]	x	x	x	x	x	x	x	x	x
SRV fails to reseat[**]							x	x	x
Operator Actions									
RCP tripped at HPI + 30 s	x	x	x	x	x	x	x	x	x
Full EFW flow after SG dryout				x	x				
Isolates SGs at 20 min						x			
Group number (PSBG-)	1	2	3	2	4	5	1	2	3

[*] Other conditions common to all cases: full power, decay heat 1.0 times ANS, all ECCS components operate as designed

[**] If challenged

[***] Break is PORV-sized and may occur anywhere

C.6.1.3 Applicable Data from Other Cases

The INEL 2-in. hot leg and 2.5-in. cold leg LOCA cases show trends in system pressure, break flows, downcomer temperatures, and steam generator secondary response similar to the LANL PORV LOCA. This suggests a consistent cooldown mechanism over all the modeled states rather than sequence specific cooldown mechanisms for each case.

C.6.1.4 Extrapolation Assumptions

A small-break LOCA involves a loss of hot primary coolant through the break with replacement of mass in the form of cold HPI fluid. This mechanism will gradually cool off the primary while maintaining high pressure. Conditions prevailing in the primary system during a PORV LOCA include relatively constant natural circulation through both loops, vent valve flow (also relatively constant), and HPI flows that balance break flows on a volumetric basis and approximately match on a mass flow basis.

Cold leg, cold leg-HPI mixture, and downcomer temperatures may be obtained from the hot leg temperature if the loop flows and vent valve flows are known. In this study, individual loop flows of 450 lb/s, total vent valve flows of 150 lb/s, and HPI flows of 118 lb/s were employed for the PORV-sized LOCA cases. These loop and vent valve flows were maintained for over 700 s in the LANL PORV LOCA case. It is assumed that these conditions will prevail throughout the sequences being evaluated.

C.6.2 Group 1 -- Cases SBLOCA1 and SBLOCA7

a. Temperature

These cases feature normal runback of main feedwater and proper operation of emergency feedwater. Upon opening of the break at the PORV, the reactor will trip and the downcomer temperature will initially rise to 572°F at 5 s, because of reduced heat transfer through the steam generators and then slowly decline because of the release of steam through the TBVs and SSRVs. At 70 s HPI injection begins and system cooldown commences, at first rather rapidly due to the large HPI flow. The proper operation of the main feedwater system will result in temperatures roughly 86°F higher than was calculated by LANL in the PORV LOCA with overfeed. Therefore, a downcomer temperature of 500°F is attained at 550 s and remains constant to 1000 s.

This momentary pause in cooldown is due to sharp decreases in HPI flows in the same period. The continuing decrease in steam generator exit temperatures is masked by this simultaneous reduction in HPI flow. After 1000 s, the cooldown model was used to extrapolate temperatures down to 390°F at 7200 s (see Table C.26 and Figure C.24).

b. Pressure

The LANL PORV LOCA case has HPI injection start at 70 s. Likewise, the INEL 2-in hot leg-break LOCA case has HPI begin at 78 s. Therefore, it is assumed that the primary depressurizes from 2180 psia to the HPI set point, 1500 psia, by 70 s. The system reaches 1000 psia at 550 s. It is assumed that the break is located in the vapor region of

Table C.26 P, T, and h profiles for PORV-sized LOCA group 1 sequences

PSBG1 .T		PSBG1 .P		PSBG1 .h	
Time(s)	T(F)	Time(s)	P(psia)	Time(s)	H(Btu/h) (ft**2F)
0.0	555.0	0.0	2180.0	0.0	2820.0
5.0	572.0	70.0	1500.0	100.0	2820.0
60.0	552.0	550.0	1000.0	150.0	210.0
550.0	500.0	850.0	1670.0	7200.0	210.0
1000.0	498.0	7200.0	1670.0		
1500.0	494.3				
2000.0	488.2				
2500.0	480.4				
3000.0	471.5				
3500.0	461.9				
4000.0	451.9				
4500.0	441.7				
5000.0	431.6				
5500.0	421.7				
6000.0	412.0				
6500.0	402.6				
7000.0	393.6				
7200.0	390.1				

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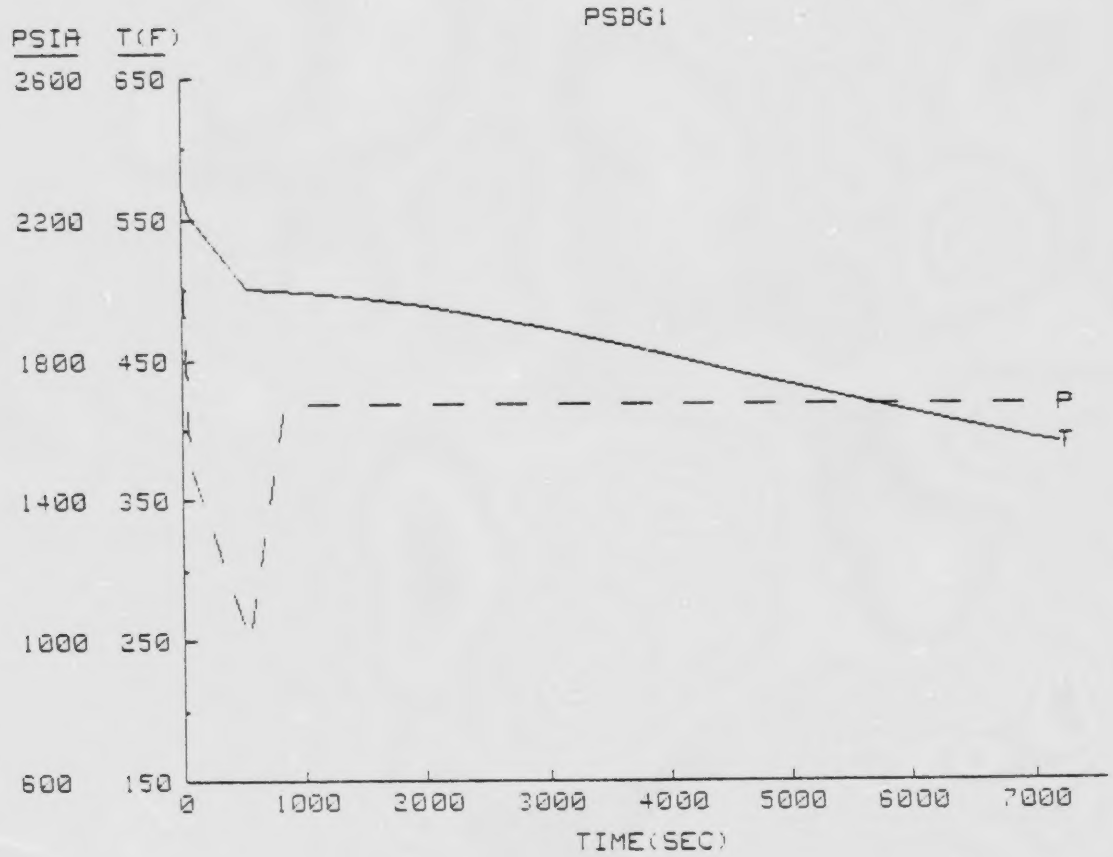

Figure C.24 Pressures and temperatures for PORV-sized LOCA group 1 sequences.

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the pressurizer. Therefore, it is necessary to go water-solid in the pressurizer to reach the final pressure of 1670 psia observed at 850 s in the LANL PORV-size LOCA. Pressure is assumed to remain at this value throughout the remainder of the sequence. The primary safety relief valves are not challenged in this sequence (see Table C.26 and Figure C.24)

c. Heat Transfer Coefficient

LANL estimated the downcomer heat transfer coefficient to 16000w/m^{2}K (2820 Btu/h ft^{2o}F) for four-RCP operation and 1200 w/m^{2}K (210 Btu/h ft^{2o}F) for natural circulation. The change between these values occurs at RCP trip at 100 s (see Table C.26).

C.6.3 Group 2 -- Cases SBLOCA2, SBLOCA4, and SBLOCA8

a. Temperature

In this group, the break is taken to be at or near the PORV in the vapor region of the pressurizer. This case follows the LANL PORV LOCA out to 1000 s because the specified overfeed resembles the overfeed in the LANL calculation. Case SBLOCA4 is included because the specified SG dryout will not occur. Thereafter, the case follows the cooldown model for the remainder of the transient. A temperature of 392.9^{o}F is projected at 7200 s (see Table C.27 and Figure C.25).

Table C.27 P, T, and h profiles for PORV-sized LOCA group 2 sequences

PSBG2 .T		PSBG2 .P		PSBG2 .h	
Time(s)	T(F)	Time(s)	P(psia)	Time(s)	H(Btu/h) (ft**2F)
0.0	555.0	0.0	2180.0	0.0	2820.0
5.0	572.0	70.0	1500.0	100.0	2820.0
60.0	552.0	550.0	1000.0	150.0	210.0
550.0	492.5	850.0	1670.0	7200.0	210.0
1000.0	490.5	7200.0	1670.0		
1500.0	487.6				
2000.0	482.5				
2500.0	475.7				
3000.0	467.8				
3500.0	459.2				
4000.0	450.2				
4500.0	441.0				
5000.0	431.7				
5500.0	422.5				
6000.0	413.5				
6500.0	404.7				
7000.0	396.2				
7200.0	392.9				

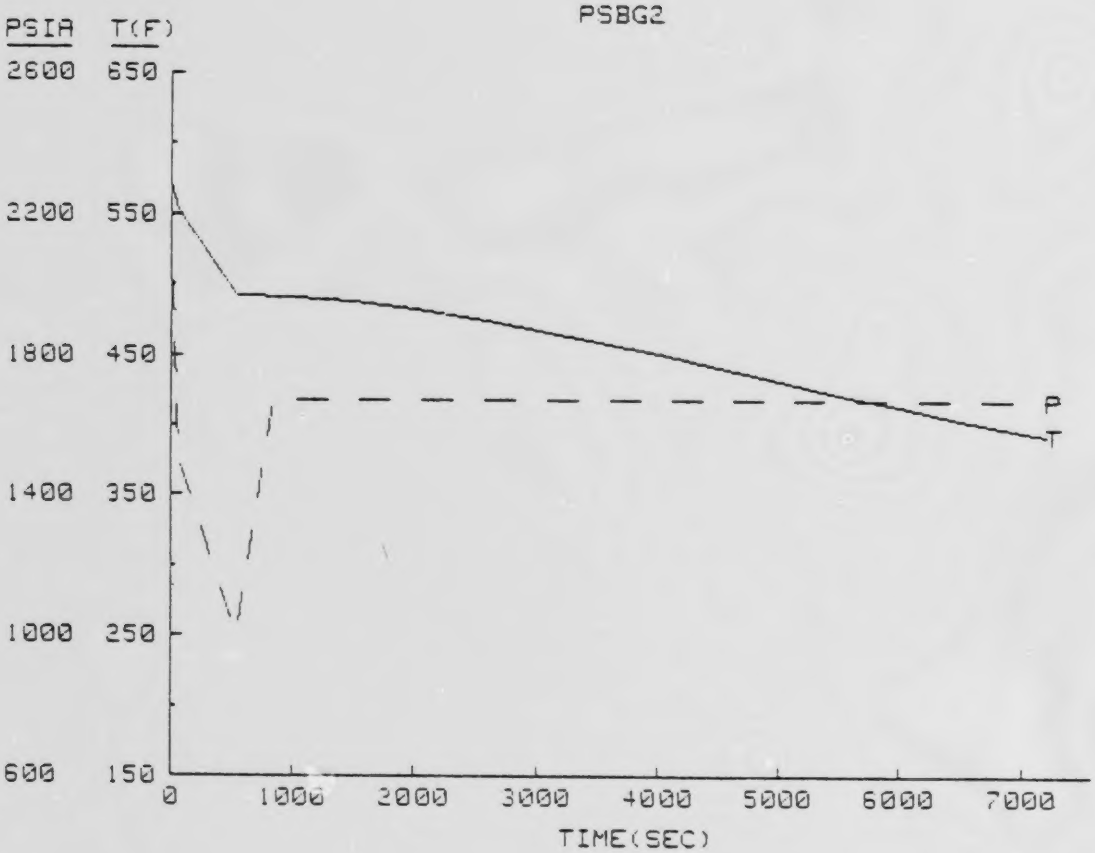

Figure C.25 Pressures and temperatures for PCRV-sized LOCA group 2 sequences.

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b. Pressure

The pressure behavior is taken to be identical with the LANL PORV LOCA calculation (see Table C.27 and Figure C.25).

c. Heat Transfer Coefficient

The heat transfer coefficients for this group are taken to be identical to group 1 (see Table C.27).

C.6.4 Group 3 -- Cases SBLOCA3 and SBLOCA9

a. Temperature

This group features loss of main feedwater at the same time the PORV-sized break opens. EFW comes on just before dryout at 50 s. Therefore, the steam generators are refilled with cold (90°F) EFW rather than warm (460°F) MFW. Assuming a total flow of 120,000 lb to fill both SGs half full, the primary temperature will be depressed from the LANL temperatures by an average of 65°F. At a rate of about 160 lb/s per loop, refill would be completed around 425 s into the sequence. The downcomer temperature would be approximately 427°F at 550 s. The temperature will rise slightly through 1500 s because of the changeover from SG feed to break flow as the principal cooldown mechanism. Then the system will cool down to 3670°F by 7200 s (see Table C.28 and Figure C.26).

Table C.28 P, T, and h profiles for PORV-sized LOCA group 3 sequences

PSBG3 .T		PSBG3 .P		PSBG3 .h	
Time(s)	T(F)	Time(s)	P(psia)	Time(s)	H(Btu/h)(ft**2F)
0.0	555.0	0.0	2180.0	0.0	2820.0
5.0	572.0	70.0	1500.0	100.0	2820.0
60.0	552.0	550.0	1000.0	150.0	210.0
550.0	427.0	850.0	1670.0	7200.0	210.0
1000.0	431.6	7200.0	1670.0		
1500.0	433.4				
2000.0	432.2				
2500.0	429.0				
3000.0	424.3				
3500.0	418.5				
4000.0	412.1				
4500.0	405.2				
5000.0	398.1				
5500.0	390.9				
6000.0	383.7				
6500.0	376.7				
7000.0	369.8				
7200.0	367.0				

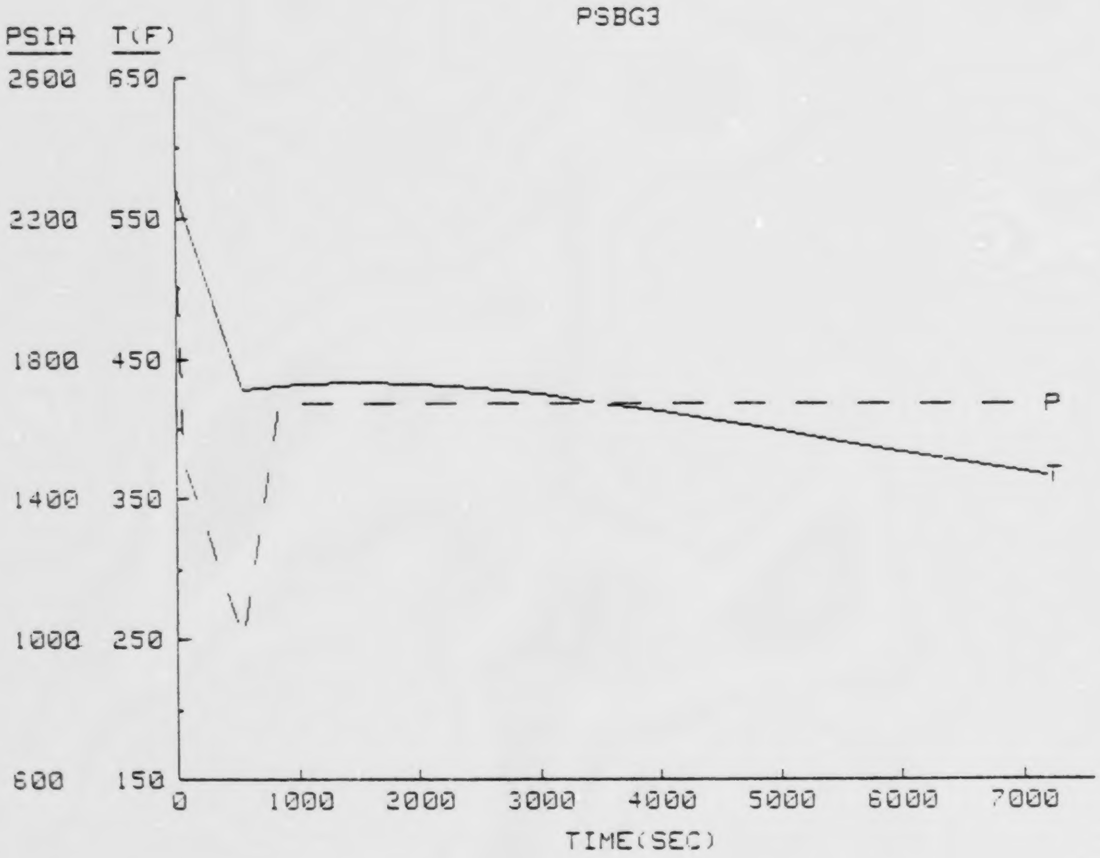

Figure C.26 Pressures and temperatures for PORV-sized LOCA group 3 sequences.

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b. Pressure

This case is taken to be identical to the LANL PORV LOCA calculation. In actual practice, the overcooling of the system by EFW will temporarily reduce system pressures. However, HPI flow would increase and sustain pressure recovery until water-solid conditions occur in the pressurizer and liquid flow choking occurs in the break (see Table C.28 and Figure C.26).

c. Heat Transfer Coefficient

This case is taken to be identical to group 1 (see Table C.28).

C.6.5 Group 4 -- Case SBLOCA5

a. Temperature

This case is characterized by a loss of main feedwater at the time the PORV-sized break opens. The steam generators boil dry and then are flooded by cold EFW, which is not controlled to level. Certain elements of this sequence have similarities to the Rancho Seco-type event. Applicable information includes EFW flows of 160 lb/s into each loop until both steam generators and steam lines are full, time of 3000 s to fill secondary systems with EFW, and decline of EFW flow rate to 120 lb/s into each loop with liquid flow through TBVs.

The temperature response of the system was projected on a piecewise basis. Through 60 s, system response is taken to be normal with the downcomer temperature at 552°F at the end of this interval.

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Between 60 s and 800 s, the EFW starts to overwhelm the system, and the steam generators completely fill by the end of this period (downcomer temperature = 384.2°F). Overfill continues at 160 lb/s per steam generator until 3000 s into the transient when the secondaries become water-solid and liquid flow restriction increases pressure and reduces flow to 120 lb/s per steam generator. (The sequence may not progress this far if the EFW turbine pump trips due to low steam quality.) The temperature at 3000 s is 239.3°F. This assumes that the EFW is heated from 90°F to the current steam generator primary exit as it passes through the steam generators. The LANL Rancho Seco-type transient calculation bears this assumption out, but it is still possible that much of the feedwater will bypass directly to the steam lines and not cool the system down. In the final mode (120 lb/s), temperature reaches 171.3°F at 7200 s. This severe overcooling is due to the compounding of HPI injection and SG overfeed cooldown mechanisms (see Table C.29 and Figure C.27).

b. Pressure

The massive overcooling will decrease system pressure below the values used previously. It is assumed that the HPI fails to "gain" on the system until the steam generators are refilled with EFW, or for 800 s. This will delay the attainment of final system pressure (1670 psia) to 1650 s (see Table C.29 and Figure C.27).

Table C.29 P, T, and h profiles for PORV-sized LOCA group 4 sequences

PSBG4 .T		PSBG4 .P		PSBG4 .h	
Time(s)	T(F)	Time(s)	P(psia)	Time(s)	H(Btu/h) (ft**2F)
0.0	555.0	0.0	2180.0	0.0	2820.0
5.0	570.0	70.0	1500.0	100.0	2820.0
60.0	552.0	550.0	1000.0	150.0	210.0
500.0	430.9	1350.0	1000.0	7200.0	210.0
800.0	394.2	1650.0	1670.0		
1000.0	373.5	7200.0	1670.0		
1500.0	328.7				
2000.0	292.5				
2500.0	263.1				
3000.0	239.3				
3500.0	226.4				
4000.0	215.2				
4500.0	205.4				
5000.0	196.9				
5500.0	189.5				
6000.0	183.2				
6500.0	177.7				
7000.0	173.0				
7200.0	171.3				

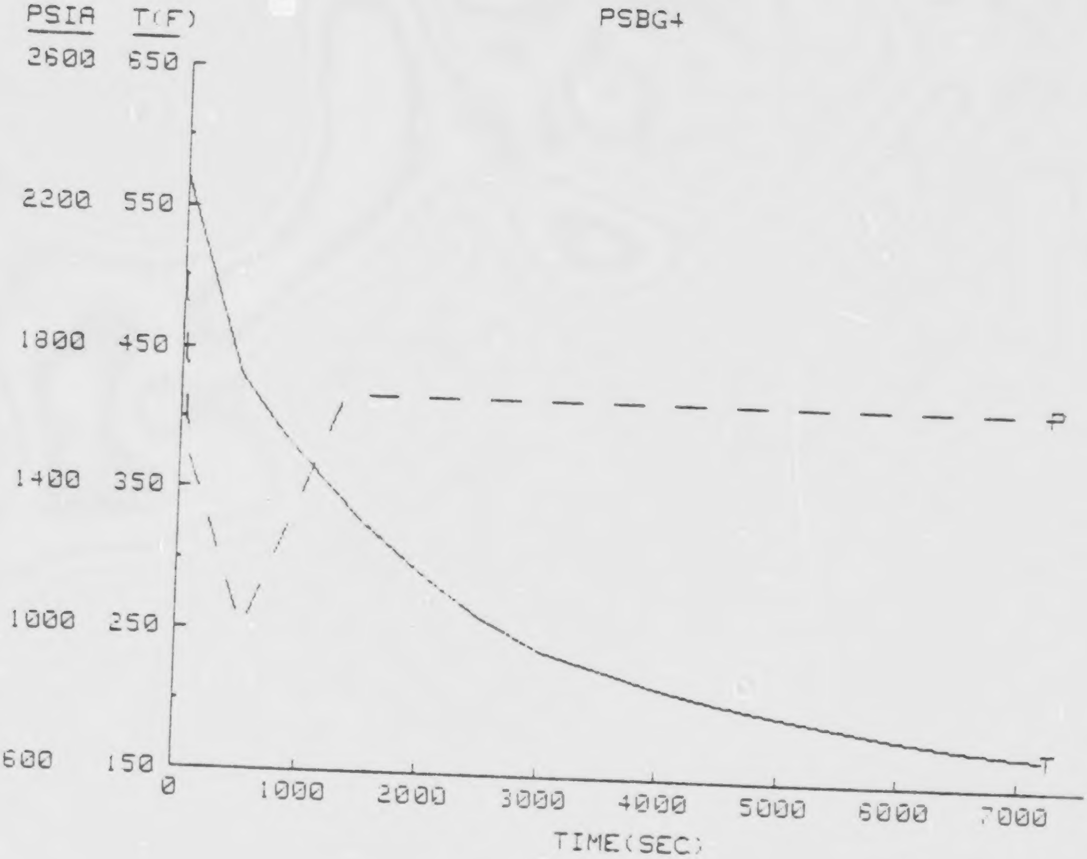

Figure C.27 Pressures and temperatures for PORV-sized LOCA group 4 sequences.

538

c. Heat Transfer Coefficient

This case is identical to group 1 (see Table C.29).

C.6.6 Group 5 -- Case SBLOCA6

a. Temperature

In this case, the PORV-sized break is complicated by failure of two TBVs in the open position on SGA. Main feedwater at 400°F is assumed available until steam generator A feedwater is isolated at 20 min. The temperature is assumed to rise from 555°F to 572°F at 5 s and then drop to 552°F as a result of TBV operation. Thereafter, the TBVs remain open, and the system temperature drops mainly because of heat loss as steam through the open TBVs. The downcomer temperature falls to 351.8°F at 1200 s where feedwater isolation occurs. Steam loss continues until affected SGs dry out at 2500 s where a minimum temperature of 311.7°F (average) occurs. If the TBVs were also isolated at 1200 s, this additional cooling would not occur. If loop B has stagnated, then minimum temperatures around 250°F will occur at this point.

With steam generator A removed as a heat sink, the system equilibrates, and downcomer temperatures rise as hot leg temperatures remain steady until HPI flowthrough becomes the principal heat removal mechanism. After rising to 342°F between 3500 and 4000 s, downcomer temperature declines to 325.6°F at 7200 s. (See Table C.30 and Figure C.28.)

Table C.30 P, T, and h profiles for PORV-sized LOCA group 5 sequences

PSBG5 .T		PSBG5 .i		PSBG5 .H	
Time(s)	T(F)	Time(s)	P(psia)	Time(s)	H(Btu/h) (ft**2F)
0.0	555.0	0.0	2180.0	0.0	2820.0
5.0	572.0	70.0	1500.0	100.0	2820.0
60.0	552.0	550.0	1000.0	150.0	210.0
200.0	423.9	1200.0	1000.0	7200.0	210.0
800.0	368.8	1500.0	1670.0		
1200.0	351.8	7200.0	1670.0		
1600.0	338.0				
2000.0	325.6				
2200.0	319.8				
2500.0	311.7				
3000.0	341.2				
3500.0	342.4				
4000.0	342.1				
4500.0	340.9				
5000.0	338.9				
5500.0	336.4				
6000.0	333.5				
6500.0	330.3				
7000.0	326.9				
7200.0	325.6				

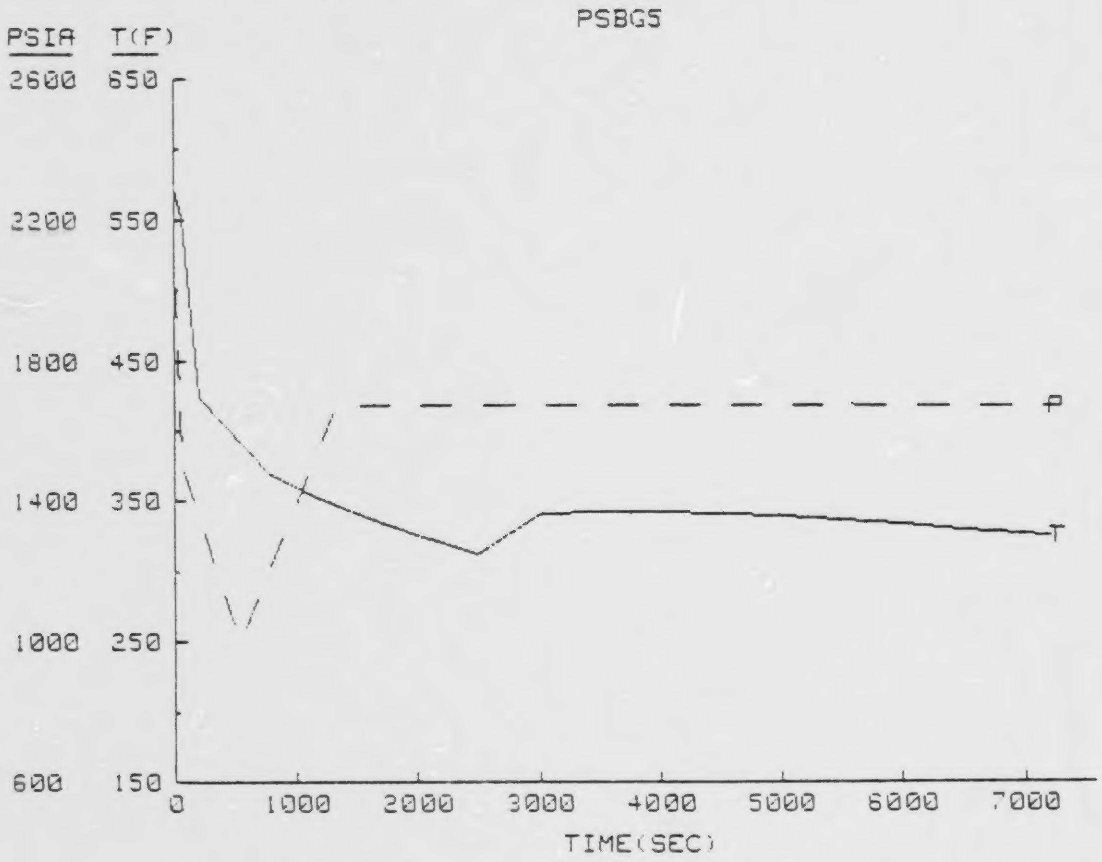

Figure C.28 Pressures and temperatures for PORV-sized LOCA group 5 sequences.

541

b. Pressure

Response is assumed to be similar to that of group 4 except that pressure begins to rise at SGA isolation (1200 s). (See Table C.30 and Figure C.28.)

c. Heat Transfer Coefficient

This case is identical to group 1 (see Table C.30).

C.7 Feedwater Transient Cases

C.7.1 ORNL-Defined Feedwater Transient Cases

The conditions for the seven ORNL-defined feedwater transient cases are described in Table C.31. Due to the variations in most of the sequences, only cases 1 and 2 were grouped for evaluation. The cases are described in detail below.

C.7.1.1 Bases

INEL maximum sustainable overfeed
LANL Rancho Seco-type transient

C.7.2.1 Departures from Bases

a. Most cases allow main feedwater trip and EFW control on level while base cases do not.

b. SG dryout cases feature extended delays in feedwater restoration (20 min) to allow heatup of vessel.

C.7.1.3 Applicable Data from Other Cases

LANL TBV cases 5A, 5C, 6A, 6C also feature overfeeds and provide feedwater temperature trend information and system behavior with RCPs operating.

C.7.1.4 Extrapolation Assumptions, Procedures, and Results

Steam generator overfeeds are sources of cooling that also increase the total system heat capacity (mass). The system therefore becomes more resistant to further temperature changes unless other cooldown mechanisms,

Table C.31 Feedwater transient cases

Condition*	Case FW- (or OVRFD-)						
	1	2	3	4	5	6	7
Main Feedwater							
SGA runback failure	x	x			x		
Both SG runback failure			x	x			
LOMFW at time = 0.0						x	x
MFWP trip high SG level	x	x	x	x			
MFWP high SG level trip failure					x		
Emergency Feedwater							
EFW works as designed	x	x	x	x	x		
EFW on manual						x	x
Operator Actions							
RCP trip at HPI + 30 s	x		x		x	x	x
Fail to trip RCP		x		x			
EFW restored full flow at 20 min						x	
MFW restored full flow at 20 min							x

---

*Conditions common to all cases: full power, decay heat 1.0 times ANS, secondary pressure control systems operate as designed, primary pressure control system operate as designed, all ECCS components operate as designed such as HPI flow, are activated. Following reactor trip, main feedwater temperature starts to decline because of loss of heater steam. For sequences where level trips of MFW occur, main feedwater temperatures stay above 350 to 400°F. Resulting system temperatures are bounded by normal operating temperatures and this feed temperature. If the overfeed uses 90°F EFW, the resulting temperatures can be much lower.

In most cases, transient evaluation will simply use the applicable portions of the base case calculations. Where extrapolation is necessary, the cooldown model is used.

C.7.2 Cases FW1 and FW2

a. Temperature

In this case, MFW runback failure occurs in SGA while SGB runback succeeds MFW pump trip on high SG level. EFW and all other systems operate as designed. Early in the transient, SSRV and TBV steam releases account for a large portion of the cooling. All of these conditions follow LANL TBV case 5A out to 60 s when the MFW pump trips on high level. High pressure in SGA prevents further flow into the SG from HW/CB pump action. The downcomer temperature at this point is 552°F. HPI does not come on and the RCPs remain on. The system is cooled to 548°F by continuing operation of the TBVs with occassional EFW flow to SGB. This final temperature is assumed to be reached by 200 s. Temperatures are documented in files FW1.T and FW2.T. This case is very similar to FW4 in which HPI comes on but the RCPs are left running. Therefore, HPI does not significantly affect downcomer temperature because of excellent mixing. If HPI comes on and the RCPs are tripped, as in case FW2, the temperatures and pressures reported in case required FW3 would be obtained (see Table C.32 and Figure C.29).

b. Pressure

Pressure is assumed to follow LANL TBV case 5A out to 60 s. The pressure declines from 2180 psia to 1760 psia over this period. With the cooling mechanism curtailed, reduction to HPI set point does not occur. With over 6 ft of water still in the pressurizer, it is assumed that the heaters are activated and accomplish repressurization to 2180 psia by 200 s (see Table C.32 and Figure C.29).

c. Heat Transfer Coefficient

Since the reactor coolant pumps remain off throughout the transient, the heat transfer coefficient is constant at 2820 Btu/h $ft^2{}^oF$ (see Table C.32).

C.7.3 Case FW3

a. Temperature

In this case the startup flow control valves remain open to both steam generators, causing main feedwater overfeed to both steam generators. Both steam generators reach the MFW trip level at about 250 s (LANL case 6A). Downcomer temperature is about 530°F at this point (from INEL overfeed). HPI trip has not yet occurred and does not occur for another 20 s in the INEL overfeed. If HPI does not come on, the system temperatures will begin to recover. To satisfy the scenario

Table C.32 P, T, and h profiles for feedwater sequences 1 and 2

FW1 .T		FW1 .P		FW1 .h	
Time(s)	T(F)	Time(s)	P(psia)	Time(s)	H(Btu/h) (ft**2F)
0.0	555.0	0.0	2180.0	0.0	2820.0
5.0	572.0	60.0	1760.0	7200.0	2820.0
60.0	552.0	200.0	2180.0		
200.0	548.0	7200.0	2180.0		
7200.0	548.0				

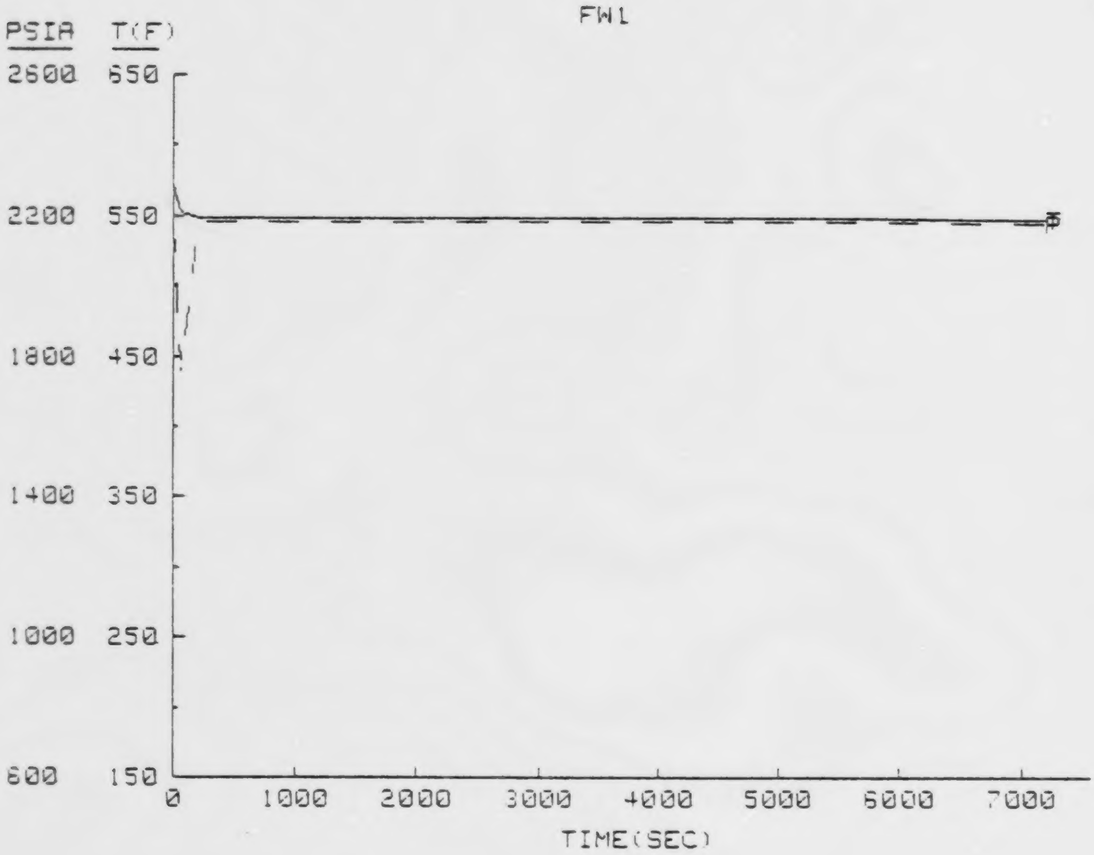

Figure C.29 Pressures and temperatures for feedwater sequences 1 and 2.

requirements, it is assumed that the HPI set point (1500 psi) is reached at 250 s and that the RCPs are tripped 30 s later. Downcomer temperatures stay around 532.7°F at 280 s, when the RCPs are tripped and then drop to 493.8°F at 350 s where natural circulation flow is established. Upon reaching the PORV set point pressure at 550 s, flow drops from an average 100 lb/s to a final value of 70 lb/s for the remainder of the transient. The reduction in cooling coupled to decay heat input raises the hot leg temperature until the TBVs actuate and the control steam generator temperature goes to 547°F. After 5000 s, decay heat has reduced to the point that the TBVs remain closed and the system cools down, because of HPI flow, to a final value of 510.2°F at 7200 s (see Table C.33 and Figure C.30).

b. Pressure

Pressure is assumed to drop to 1500 psia at 250 s to bring about HPI injection. The HPI is assumed to restore the system to the PORV set point pressure 2430 psia at 550 s. Pressure remains at this value for the rest of the sequence (see Table C.33 and Figure C.30).

c. Heat Transfer Coefficient

Heat transfer coefficient drops from 2180 Btu/h/ft^{2}°F at RCP trip (250 s) to 210 Btu/h/ft^{2}°F after pump rundown at 300 s and remains at this value for the remainder of the sequence (see Table C.33).

549

Table C.33 P, T, and h profiles for feedwater sequence 3

FW3 .T		FW3 .P		FW3 .h	
Time(s)	T(F)	Time(s)	P(psia)	Time(s)	H(Btu/h) (ft**2F)
0.0	555.0	0.0	2180.0	0.0	2820.0
5.0	572.0	250.0	1500.0	250.0	2820.0
60.0	552.0	550.0	2430.0	300.0	210.0
250.0	531.1	7200.0	2430.0	7200.0	210.0
280.0	532.7				
350.0	493.8				
500.0	496.9				
750.0	515.7				
5000.0	515.7				
7200.0	510.2				

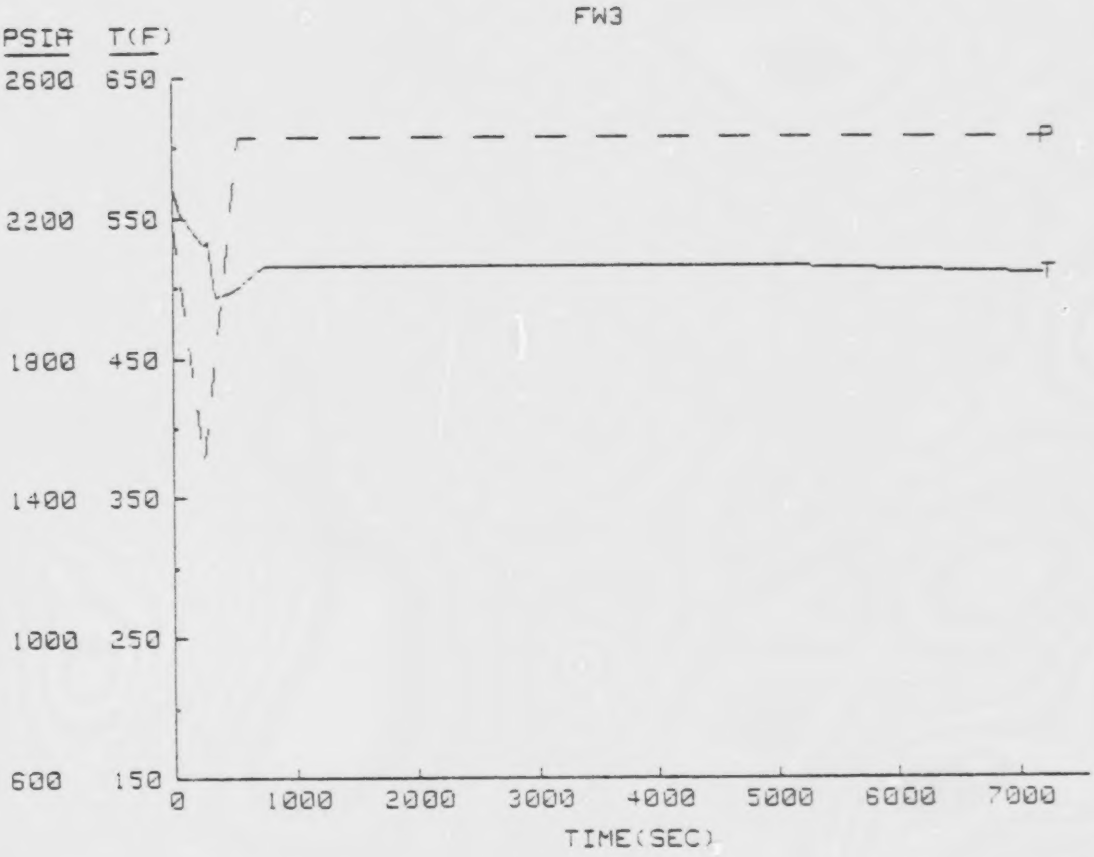

Figure C.30 Pressures and temperatures for feedwater sequence 3.

551

C.7.4 Case FW4

a. Temperature

This case is similar to case FW3 except that the operator fails to trip the RCPs after HP1 initiation. The extensive mixing and additional heat input will cause the temperature to recover from 532°F at 280 s to 547°F at about 400 s where it is controlled by TBV action for the rest of the sequence (see Table C.34 and Figure C.31).

b. Pressure

This case is taken to be identical to case FW3 (see Table C.34 and Figure C.31).

c. Heat Transfer Coefficient

Since the RCPs are not tripped, this case is taken to be identical to case FW1 (see Table C.34).

C.7.5 Case FW5

a. Temperature

This case features an overfeed to SGA where the high SG level MFW pump trip also fails. SGB feedwater is properly controlled by ICS. SGA receives approximately 1500 lb/s of MFW in this case, compared to the 300 lb/s when both SGs are being overfed. Therefore, the time to MFW trip on low steam quality in the steam line will be dropped from 417 s in the INEL overfeed case to approximately 85 s. The temperature response is assumed to be similar to case FW1 because,

Table C.34 P, T, and h profiles for feedwater sequence 4

FW3 .T		FW3 .P		FW3 .h	
Time(s)	T(F)	Time(s)	P(psia)	Time(s)	H(Btu/h)(ft**2F)
0.0	555.0	0.0	2180.0	0.0	2820.0
5.0	572.0	250.0	1500.0	7200.0	2820.0
60.0	552.0	600.0	2430.0		
250.0	530.0	7200.0	2430.0		
280.0	535.6				
600.0	548.0				
7200.0	548.0				

553

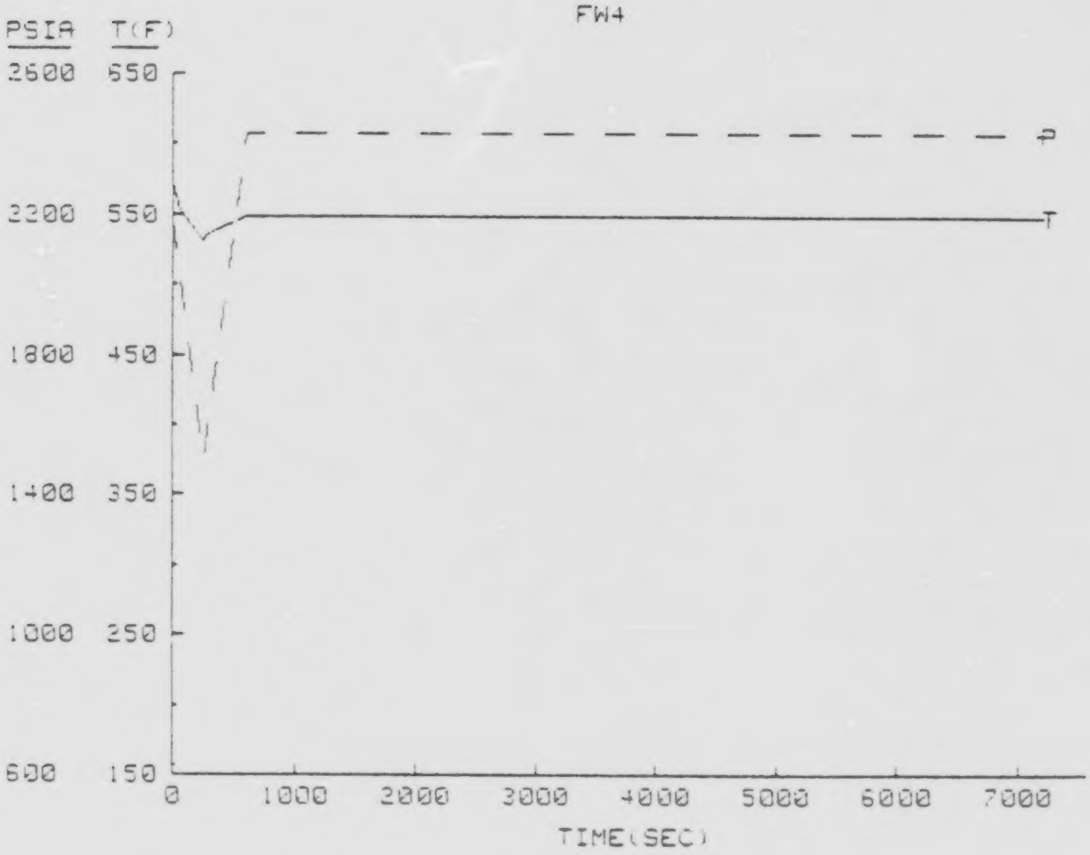

Figure C.31 Pressures and temperatures for feedwater sequence 4.

after the steam generator is full, additional feedwater will only displace other feedwater that has not had time to absorb heat. The minimum temperature is 548°F at 85 s. The HPI system does not come on and therefore the RCPs are not tripped. Even if these trips had occurred, the temperatures would remain at or above the final value of 510°F obtained for case FW3 (see Table C.35 and Figure C.32).

b. Pressure

Pressure is assumed to decline at the same rate as observed in case FW1. At 85 s, the pressure would be 1585.0 psia. This is quite close to the HPI set point and may possibly be reached in a transient of this type. However, it is assumed that the HPI does not come on and that the pressurizer heaters are able to restore the system to 2180 psia by 285 s (see Table C.35 and Figure C.32).

c. Heat Transfer Coefficient

Since the RCPs are not tripped, the heat transfer coefficient remains at its initial value of 2180 Btu/h/ft^2°F throughout the transient in case FW1 (see Table C.35).

C.7.6 Case FW6

a. Temperature

This case features steam generator dryout with a 20-min delay until EFW is restored at full flow. EFW is not controlled to level. HPI comes on as designed, and the RCPs are tripped 30 s after HPI initiation. The early portions of this transient are very similar to the Rancho

Table C.35 P, T, and h profiles for feedwater sequence 5

FW5 .T		FW5 .P		FW5 .h	
Time(s)	T(F)	Time(s)	P(psia)	Time(s)	H(Btu/h) (ft**2F)
0.0	555.0	0.0	2180.0	0.0	2820.0
5.0	572.0	60.0	1760.0	7200.0	2820.0
60.0	552.0	85.0	1585.0		
85.0	548.0	285.0	2180.0		
7200.0	548.0	7200.0	2180.0		

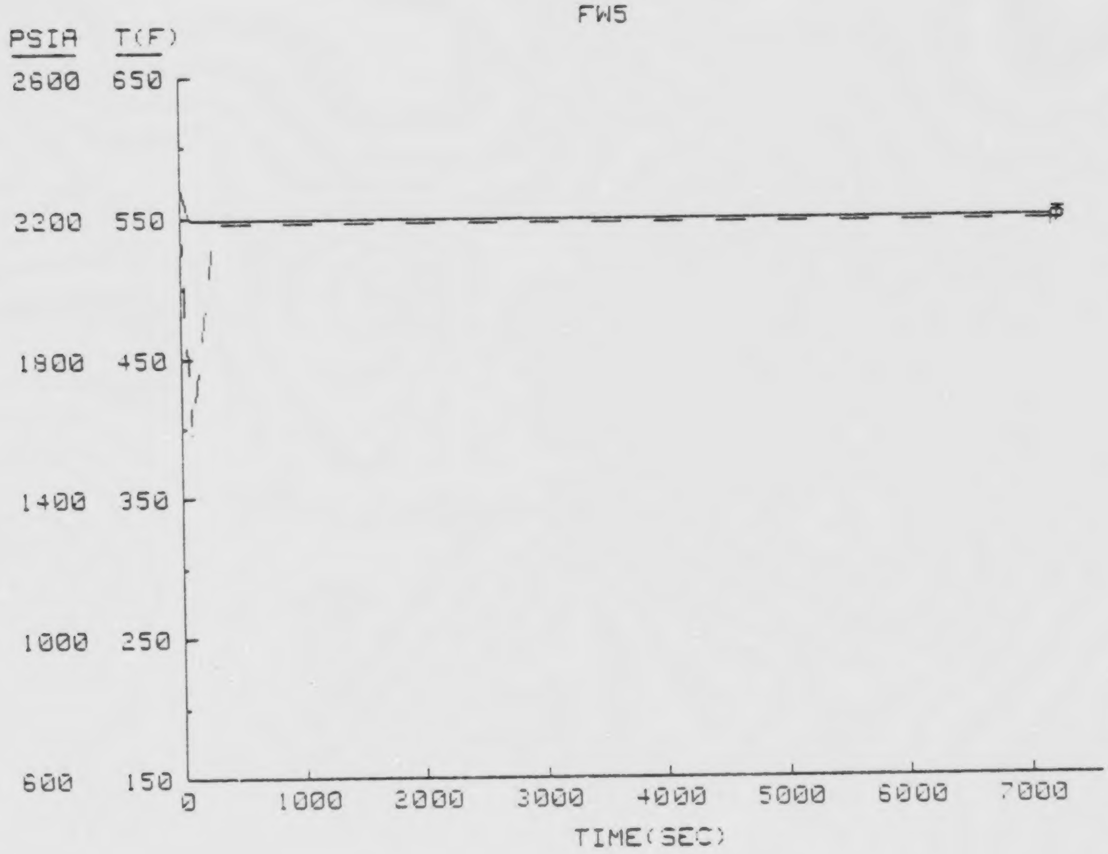

Figure C.32 Pressures and temperatures for feedwater sequence 5.

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Seco-type event. In following this sequence, the downcomer temperature would rise from its initial value of 555°F to 581°F at 8 s and then to 644°F at 1200 s ($dT/dt=+0.0524°F/s$) This temperature is 20°F below the saturation temperature for the PORV set point pressure. The introduction of cold (90°F) EFW at 1200 s will cause a decline ($dT/dt=0.228°F/s$) to 548°F at 1535 s when the RCPs are tripped. (In the LANL Rancho Seco-type transient, the temperature at HPI = 30 s is 548°F. It is assumed that cooling the system to this state will yield comparable HPI response.) After the pumps spin down, the system is assumed to attain natural circulation flows on the order of 820 lb/s per loop and total vent valve flows of 45 lb/s, as observed in LANL TBV cases 6A, 6B, and 6C under similar high cooldown rate circumstances. In this regime, the downcomer temperature decreases to 424°F at 2200 s when the steam generators are liquid full, and the HPI flow decreases as the primary is nearing the PORV set point pressure. The temperature at 7200 s is 295°F, if all flows continue as stated. However, it is likely that EFW will be lost at some earlier point because of low steam quality in the secondary, impacting the operation of the turbine-driven pump. As a result, the temperatures after the loss of EFW would be higher than the values presented here (see Table C.36 and Figure C.33).

b. Pressure

Pressure is generally taken to mimic the Rancho Seco-type event calculated by LANL. From an initial value of 2180 psia, the pressure increases to the PORV set point, 2430 psia, at 226 s and remains until EFW activation at 1200 s. (Early dips and peaks in the LANL

Table C.36 P, T, and h profiles for feedwater sequence 6

FW6 .T		FW6 .P		FW6 .h	
Time(s)	T(F)	Time(s)	P(psia)	Time(s)	H(Btu/h) (ft**2F)
0.0	555.0	0.0	2180.0	0.0	2820.0
8.0	581.0	226.0	2430.0	1540.0	2820.0
1320.0	644.0	1320.0	2430.0	1590.0	210.0
1540.0	548.0	1510.0	1500.0	7200.0	210.0
2200.0	423.7	2300.0	2430.0		
3000.0	398.5	7200.0	2430.0		
4000.0	369.5				
5000.0	343.3				
6000.0	319.9				
7000.0	299.1				
7200.0	295.2				

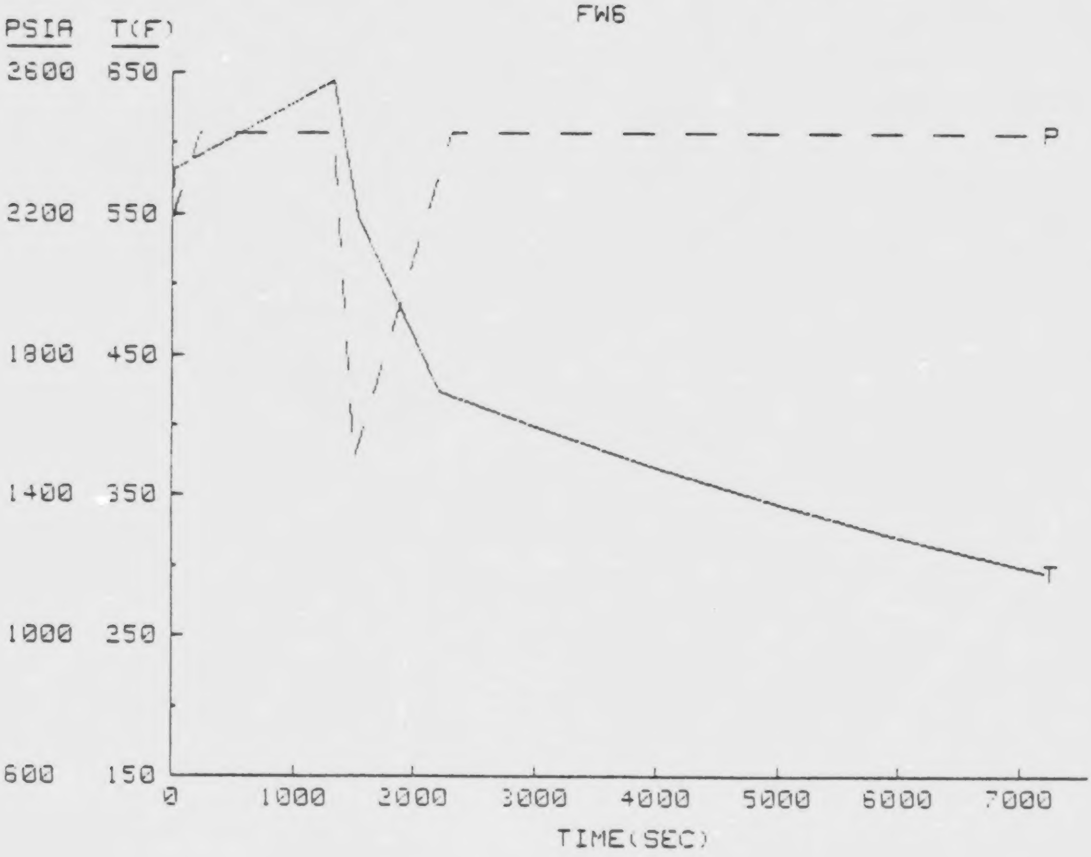

Figure C.33 Pressures and temperatures for feedwater sequence 6.

calculation were not included because of the high system temperatures.) Pressure drops to 1500 psia at 1510 s where HPI comes on to reserve the trend. Repressurization to the final pressure, 2430 psia, occurs at 2300 s (see Table C.36 and Figure C.33).

c. Heat Transfer Coefficient

The downcomer heat transfer coefficient retains its initial value, 2820 $Btu/h/ft^{2o}F$, to 1540 s when the RCPs are tripped. After a 50-s coast down period, the heat transfer coefficient obtains its final value of 210 $Btu/h/ft^{2o}F$ (see Table C.36).

C.7.7 Case FW7

a. Temperature

This case is similar to FW6 except that the steam generators are refilled with MFW at full flow starting at 20 min. Temperature behavior to 1200 s is the same as case FW6. A flow of 750 lb/s of 450^oF water to each steam generator reduces downcomer temperature to 545^oF at 1280 s when the RCPs are assumed to be tripped. The flow path switches over to the EFW headers, and flow is limited to the SUFCV value of 112.5 lb/s per steam generator. The steam generators fill, and the MFW trip on low steam line quality is assumed to occur at 2100 s with a downcomer temperature of 506^oF. With HPI flowthrough as the only cooldown mechanism, the temperature rises to 528^oF at 5000 s and declines to 523^oF at 7200 s (see Table C.37 and Figure C.34).

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Table C.37 P, T, and h profiles for feedwater sequence 7

FW7 .T		FW7 .P		FW7 .h	
Time(s)	T(F)	Time(s)	P(psia)	Time(s)	H(Btu/h) (ft**2F)
0.0	555.0	0.0	2180.0	0.0	2820.0
8.0	581.0	226.0	2430.0	1280.0	2820.0
1200.0	644.0	1200.0	2430.0	1330.0	210.0
1280.0	545.0	1250.0	1500.0	7200.0	210.0
1900.0	510.3	2100.0	2430.0		
2100.0	506.5	7200.0	2430.0		
3000.0	522.0				
4000.0	527.5				
5000.0	528.8				
6000.0	527.3				
7000.0	524.2				
7200.0	523.4				

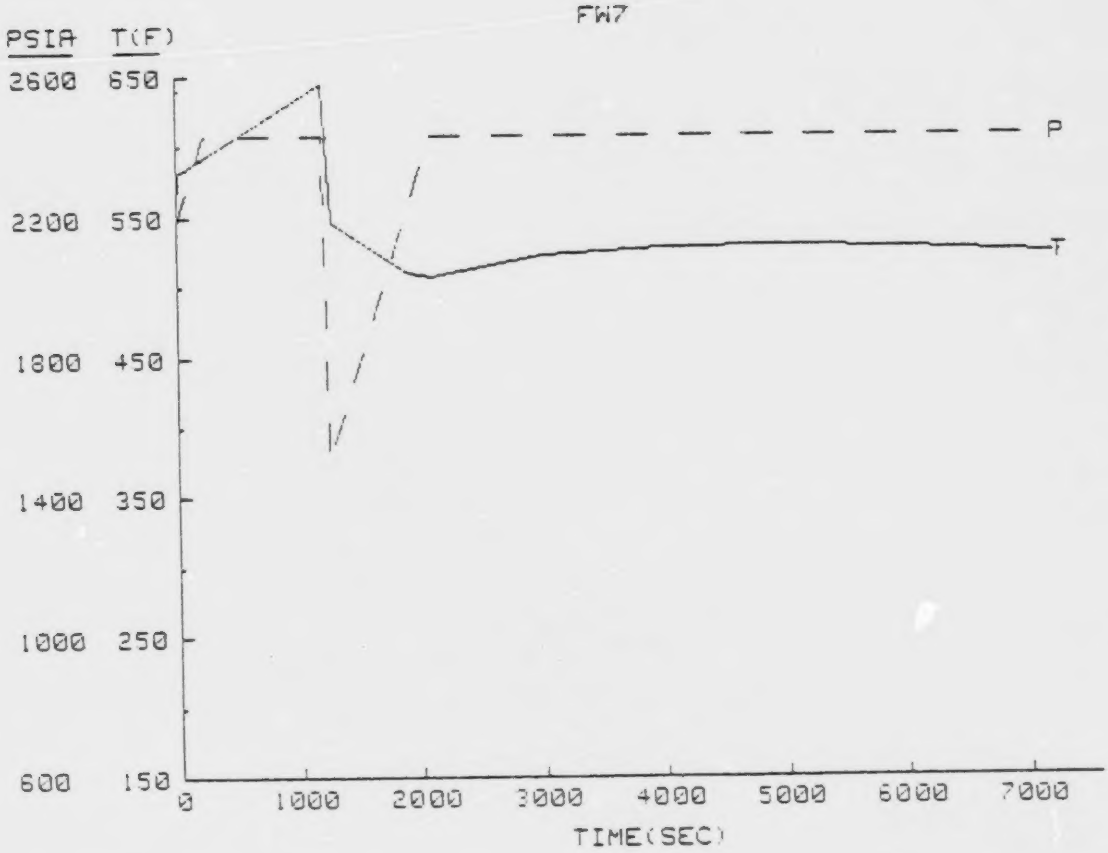

Figure C.34 Pressures and temperatures for feedwater sequence 7.

b. Pressure

Pressures follow case FW6 to 1200 s. The massive in-rush of MFW and subsequent primary heat removal is taken to drop the pressure to the HPI set point, 1500 psia, at 1250 s. The HPI is assumed to check further depressurization and begin repressurization, although it is possible that pressure may yet decrease because of continued overcooling. The PORV set point pressure is assumed to be reached at 2100 s where it remains for the duration of the sequence (see Table C.37 and Figure C.34).

c. Heat Transfer Coefficient

The heat transfer coefficient is assumed to remain constant at its initial value, 2820 $Btu/h/ft^{2o}F$, until RCP trip at 1280 s, when it ramps down to its final value of 210 $Btu/h/ft^{2o}F$ by 1330 s (see Table C.37).

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C.8 Steam Generator Tube Rupture

A steam generator tube rupture is a form of small-break LOCA. The case prepared by INEL was digitized and appears in Table C.38 and Figure C.35. This case does not appear worthy of concern from PTS standpoint, largely because of the smallness of the break, which withdraws less flow (40 lb/s) than the HPI injects at the PORV set point pressure (70 lb/s). Guillotine breaks of two tubes would give a break size comparable to a stuck-open PORV. The temperatures, pressures, and heat transfer coefficients for case PORV-sized LOCA group 11 may be representative of a two-tube rupture if flow complexities do not greatly alter the results. Larger, multitube rupture incidents would tend to resemble larger LOCAs, although void formation in the upper hand and candy canes and similar effects would greatly complicate the sequence.

Table C.38 P, T, and h profiles for single-tube steam generator tube rupture case

SGTR .T		SGTR .P		SGTR .h	
Time(s)	T(F)	Time(s)	P(psia)	Time(s)	H(Btu/h) (ft**2F)
0.0	557.0	0.0	2180.0	0.0	2820.0
320.0	563.0	160.0	2110.0	975.0	2820.0
385.0	552.7	320.0	2090.0	1025.0	210.0
625.0	552.7	366.0	1640.0	7200.0	210.0
795.0	557.0	410.0	1610.0		
940.0	552.7	650.0	1610.0		
1040.0	458.0	720.0	1640.0		
1760.0	450.0	925.0	1509.0		
2240.0	469.0	1135.0	1610.0		
7200.0	477.0	1350.0	1610.0		
		1550.0	1815.0		
		2200.0	2430.0		
		7200.0	2430.0		

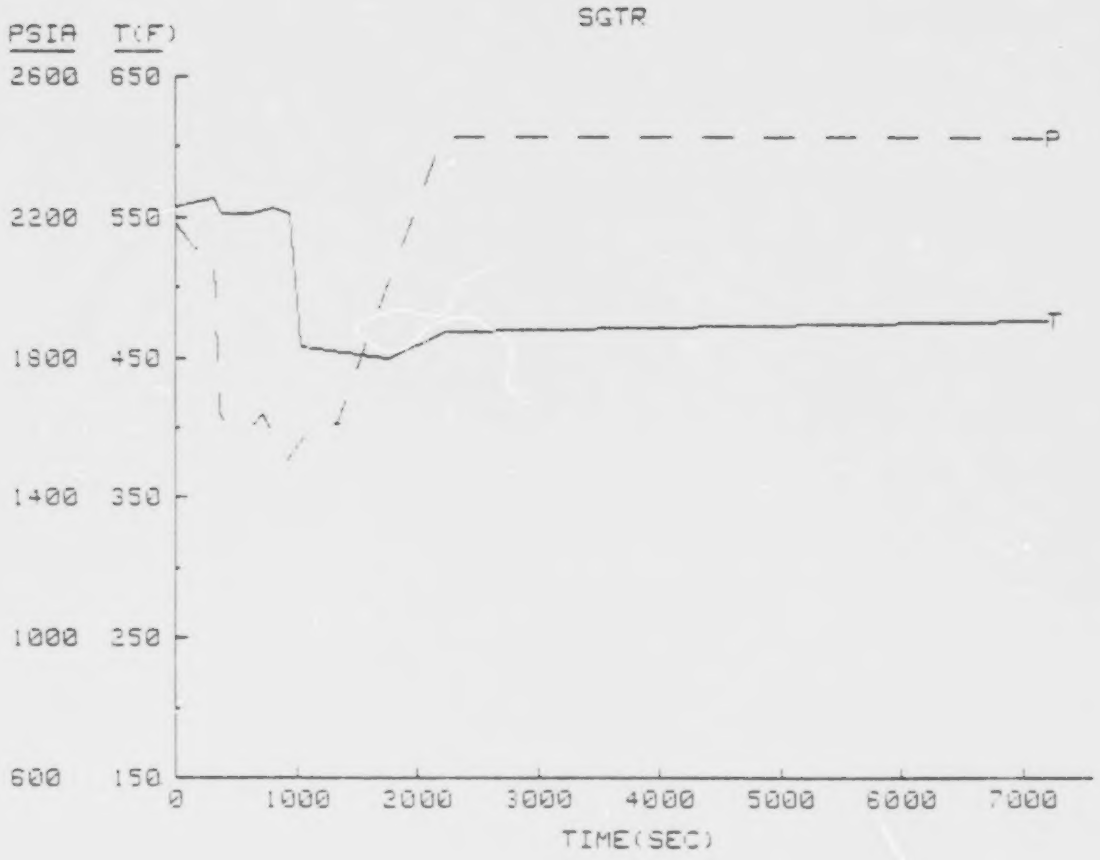

Figure C.35 Pressures and temperatures for single tube steam generator tube rupture case.

APPENDIX D

CONTRIBUTION TO P(F|E) OF FLAWS IN THE CIRCUMFERENTIAL WELDS AND THE BASE MATERIAL

Flaws anywhere in the beltline region of the reactor vessel will contribute to the probability of vessel failure. However, aside from the effect of flaw depth, some contribute more than others because of differences in orientation, length, local chemistry of the material, and local fluence. Axial flaws have the highest values of K_I, and in Oconee-1 the axial welds have higher concentrations of copper than the circumferential welds and the base material.

Because the difference in copper concentration between the axial welds and base material is rather large in terms of radiation damage, the extended surface length of an axial flaw in a weld tends to be limited to the height of a shell course, and for deep flaws this limit on surface length results in significantly lower K_I values than for much longer flaws. The surface length of extended flaws in the circumferential welds and in the base material are not limited by this mechanism, although the lengths of these flaws may be limited by gradients in fluence and coolant temperature.

Thus far our fracture-mechanics model is not sufficiently sophisticated to consider gradients in fluence and coolant temperature along the specified surface flaw path. In lieu of considering this sort of detail, all flaws in the circumferential welds and in the base material were assumed to be two dimensional. It was believed that even under these conservative conditions flaws in the circumferential welds and in the base material would not be dominant in terms of the calculated probability of vessel failure.

To be certain that the above assumption was reasonable, a few comparison calculations were made for circumferential flaws in the circumferential welds, for axial flaws in the base material, and for axial flaws in the axial welds. The flaw density was assumed to be the same for the three categories of flaws considered, and since the volume of the base material is much greater than that of the welds, the base material contributed many more flaws than the welds.

The chemistry, maximum fluences, and initial values of RTNDT for each distinct area of the vessel considered for the three categories of flaws are given in Table 5.2. In an attempt to account for the azimuthal variation in fluence in the base plate without dividing the plate region into several zones, each with a different fluence, an average fluence was used in accordance with the data in Table 5.2 and Reference 1. Also, the initial values of RTNDT given in Table 5.2 for the base material were reduced by 14°C to account for a lower radiation damage rate in the base material than in the welds.[2] The radiation-damage trend curve [ΔRTNDT = f(F, Cu, Ni)] in the fracture-mechanics model corresponds to weld material only.

The comparison calculations were made for transient No. 44 and for 32 EFPY. The results indicate that the circumferential flaws add ~5% to P(F|E) and the base material ~30%. Since the contributions were so small, flaws in the base material and the circumferential welds were not included in the remainder of the studies.

References

1. "Reactor Vessel Pressurized Thermal Shock Evaluation," DPC-RS-1001, Duke Power Company Oconee Nuclear Station, January 1982.

2. P. N. Randall, U.S. Nuclear Regulatory Commission, personal communication to R. D. Cheverton, Oak Ridge National Laboratory.

APPENDIX E

COMPILATION OF RESULTS OF OCONEE-1 PROBABILISTIC FRACTURE-MECHANICS ANALYSIS

Detailed results of the Oconee-1 probabilistic fracture-mechanics analysis are included in this appendix so that a more thorough understanding of the effect of the various assumptions used in the fracture-mechanics model and as related to input to the fracture-mechanics analysis can be obtained. For instance, the duration of all postulated transients for this study was specified as two hours. In many cases most of the failures do not occur until late in the transient. If the duration of the transient were one instead of two hours, $P(F|E)$ would be reduced substantially (∿two orders of magnitude for transient No. 44).

Sets of data are included in this appendix for each of the transients for which $P(F|E) > 10^{-7}$. A set of data includes, in this order, (1) a figure of primary-system pressure, downcomer coolant temperature, and fluid-film heat-transfer coefficient vs time in transient; (2) a summary of digital output that includes $P(F|E)$ for each weld and the vessel, the estimated error in $P(F|E)$, and histogram data for crack depths, time of failure (TWC propagation), times and values of T - RTNDT at the crack tip corresponding to initiation, arrest and failure events; (3) a plot of vessel wall temperature vs a/w, t; (4) a plot of vessel wall temperature vs t, a/w; and (5) a set of critical-crack-depth curves obtained using -2σ values of K_{Ic} and K_{Ia}, mean values of all other parameters, and fluences corresponding to 32 EFPY. The various curves in the set of critical-crack-depth curves are identified in Figure 5.17 (see Chapter 5).

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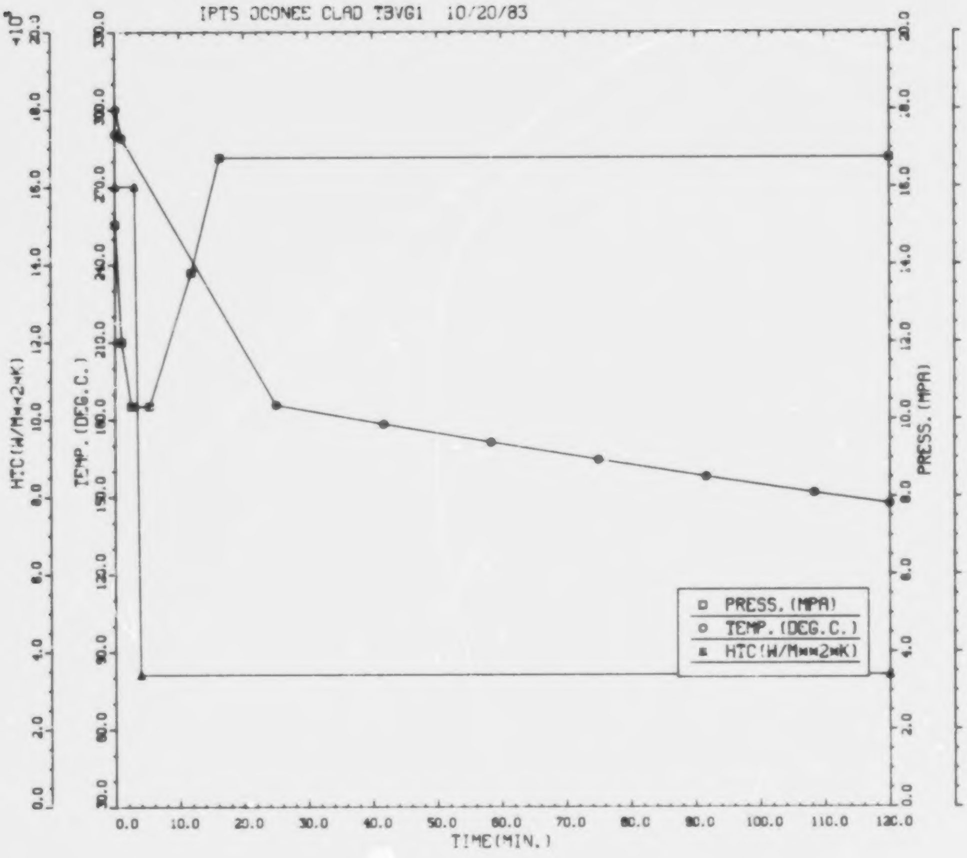
TRANSIENT NO. 1 (TBVG1)


IPTS OCONEE CLAD TBVG1 10/20/83
$\times 10^{3}$
HTC(W/M**2*K)
0.0
2.0
4.0
6.0
8.0
10.0
12.0
14.0
16.0
18.0
20.0
TEMP. (DEG.C.)
30.0
60.0
90.0
120.0
150.0
180.0
210.0
240.0
270.0
300.0
330.0
PRESS. (MPA)
0.0
2.0
4.0
6.0
8.0
10.0
12.0
14.0
16.0
18.0
20.0
TIME(MIN.)
0.0
10.0
20.0
30.0
40.0
50.0
60.0
70.0
80.0
90.0
100.0
110.0
120.0
□ PRESS. (MPA)
○ TEMP. (DEG.C.)
✱ HTC(W/M**2*K)

```
IPTS OCONEE CLAD TBVC1  10/20/83                        1. FLAWS/M**3     FO =  1.090D+19

       -------------UNADJUSTED--------------            ---ADJUSTED----
WELD    P(F/E)        95%CI     %ERR   P(INITIA)    N*V     P(F/E)    %ERR     NTRIALS


  1     4.70D-06    4.60D-06   98.00    4.70D-06  0.040   1.88D-07              500000
  2     1.17D-06    2.30D-06  196.00    1.17D-06  0.040   4.70D-08              500000
  3     7.05D-06    5.64D-06   80.02    7.05D-06  0.010   7.05D-08              500000

                                          VESSEL        3.05D-07  69.91



 DEPTHS FOR INITIAL INITIATION (MM)
                   3.2   6.3  12.7  19.0  25.4  38.1  50.8
   NUMBER           0     3     5     1     2     0     0
   PERCENT         0.0  27.3  45.5   9.1  18.2   0.0   0.0




TIMES OF FAILURE(MINUTES)
                   0.0  10.0  20.0  30.0  40.0  50.0  60.0  70.0  80.0  90.0 100.0 110.0 120.0
   NUMBER               0     0     0     0     0     0     0     0     0     1     3     7
   PERCENT            0.0   0.0   0.0   0.0   0.0   0.0   0.0   0.0   0.0   9.1  27.3  63.6

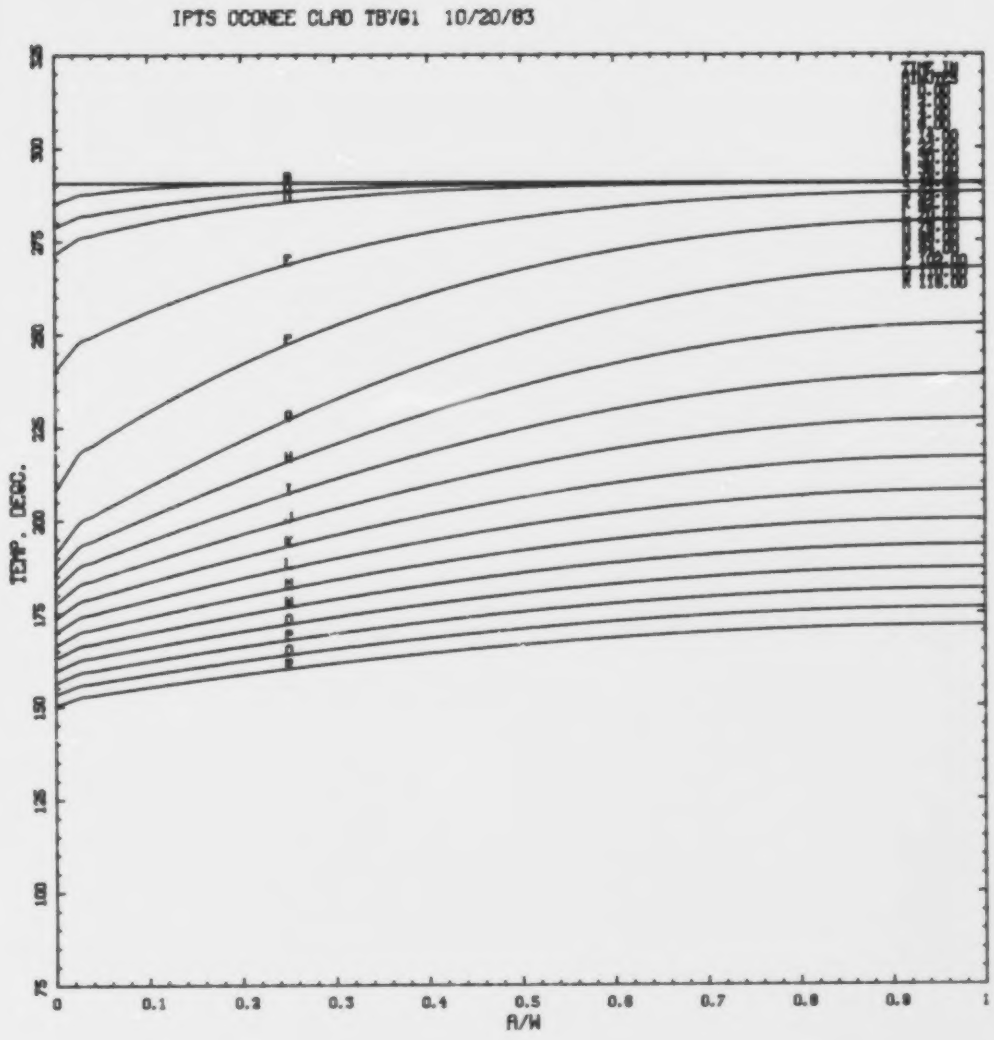


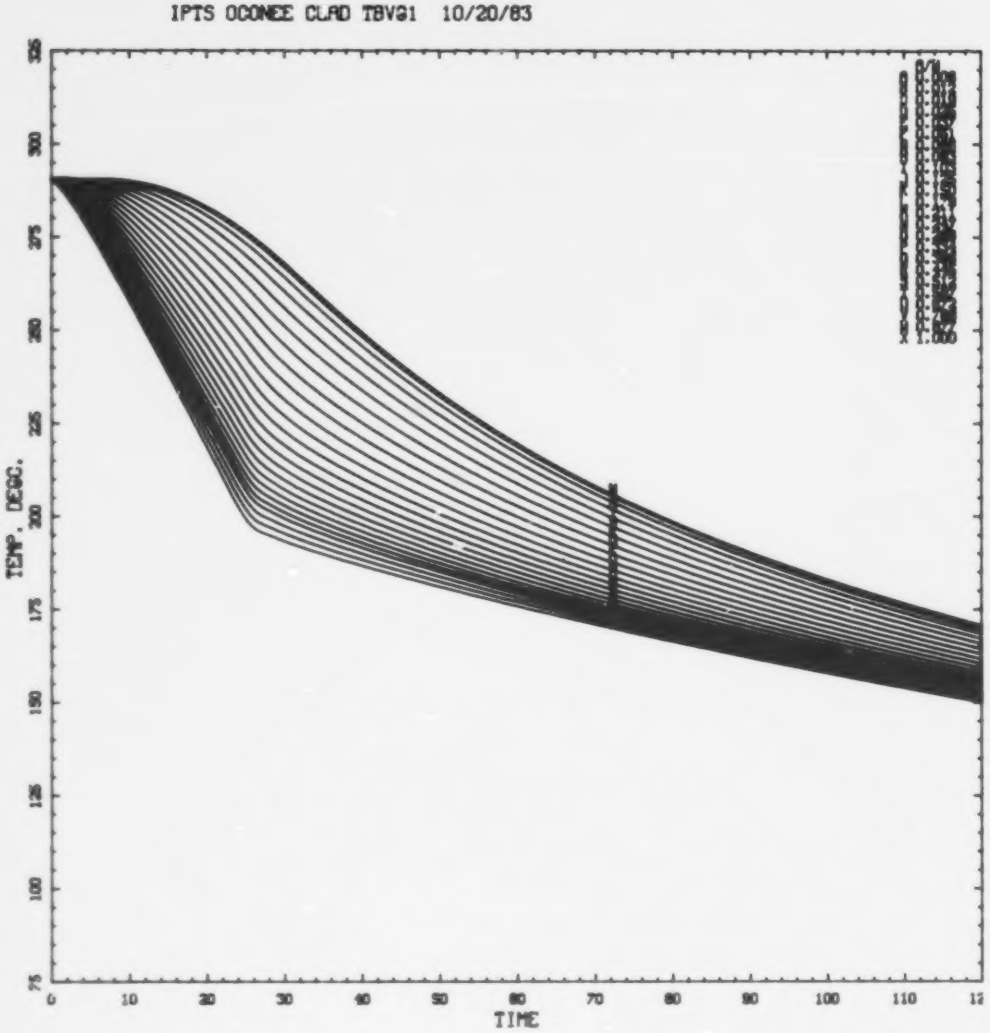

 INITIATION T-RTNDT(DEG.C)
                -55.6 -41.7 -27.8 -13.9   0.0  13.9  27.8  41.7  55.6  69.4  83.3  97.2 111.1
   NUMBER               0     0     1     7     3     1     0     0     0     0     0     0
   PERCENT            0.0   0.0   8.3  58.3  25.0   8.3   0.0   0.0   0.0   0.0   0.0   0.0



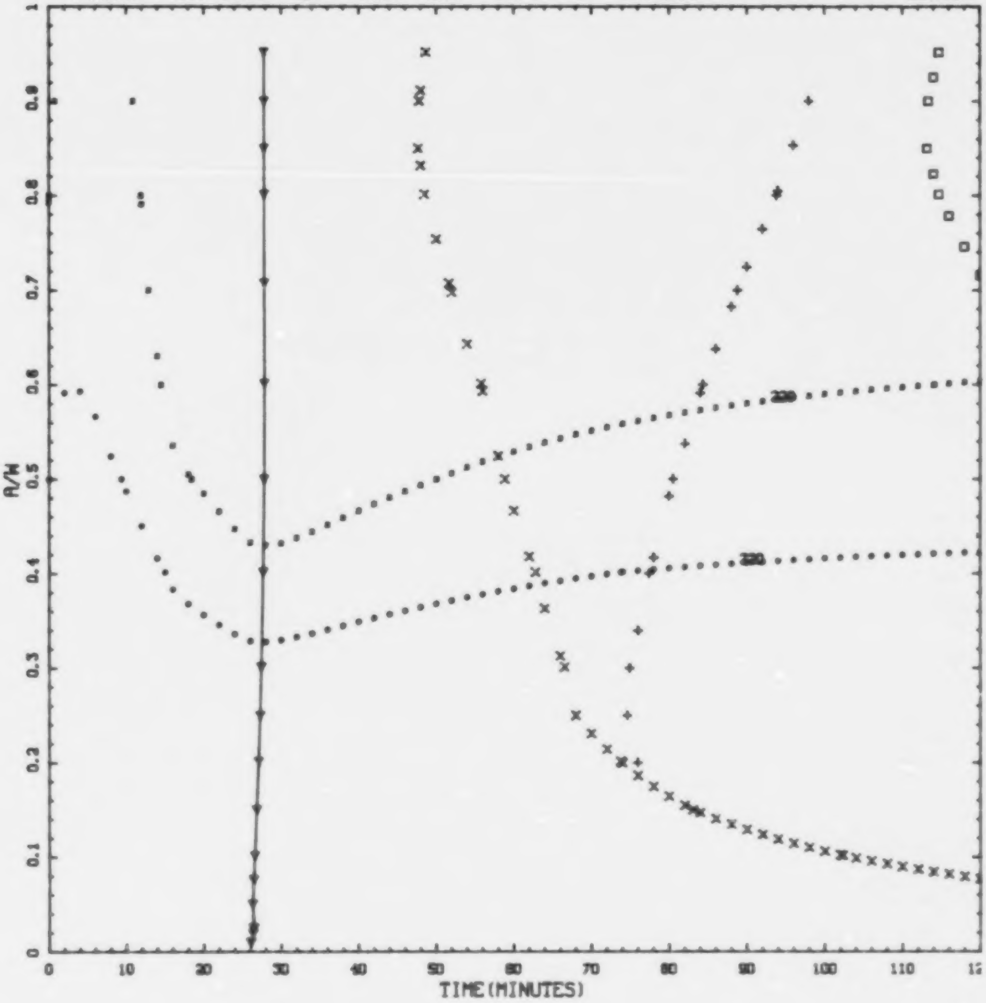

 ARREST T-RTNDT(DEG.C)
                -27.8 -13.9   0.0  13.9  27.8  41.7  55.6  69.4  83.3  97.2 111.1 125.0 138.9
   NUMBER               0     1     0     0     0     0     0     0     0     0     0     0
   PERCENT            0.0 100.0   0.0   0.0   0.0   0.0   0.0   0.0   0.0   0.0   0.0   0.0
```

AVAILABLE


IPTS OCONEE CLAD TB/01 10/20/83
TEMP. DEGC.
325
300
275
250
225
200
175
150
125
100
75
0
0.1
0.2
0.3
0.4
0.5
0.6
0.7
0.8
0.9
1
A/W
R 110.00

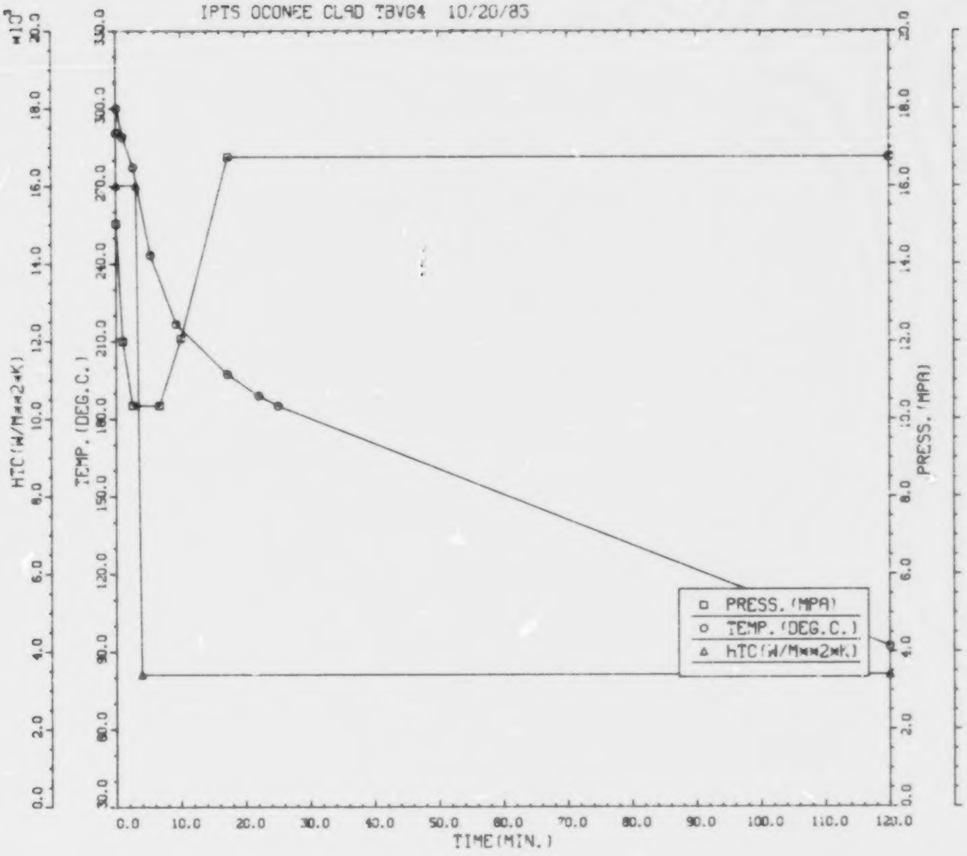

IPTS OCONEE CLAD TBVG1 10/20/83
TEMP. DEGC.
TIME
75
100
125
150
175
200
225
250
275
300
325
0
10
20
30
40
50
60
70
80
90
100
110
12

576



CRITICAL CRACK DEPTH CURVES FOR IPTS OCONEE CLAD TBV61 10/20/83
RTNDTO --6.7 DEGC %CU - 0.29 %NI - 0.55 FO - 1.09E19 LONGIT

TRANSIENT NO. 4 (TBVG4)


IPTS OCONEE CL9D TBVG4 10/20/83
HTC(W/M**2*K) ×10³
0.0
2.0
4.0
6.0
8.0
10.0
12.0
14.0
16.0
18.0
20.0
TEMP.(DEG.C.)
30.0
60.0
90.0
120.0
150.0
180.0
210.0
240.0
270.0
300.0
330.0
PRESS.(MPA)
0.0
2.0
4.0
6.0
8.0
10.0
12.0
14.0
16.0
18.0
20.0
TIME(MIN.)
0.0
10.0
20.0
30.0
40.0
50.0
60.0
70.0
80.0
90.0
100.0
110.0
120.0
□ PRESS.(MPA)
○ TEMP.(DEG.C.)
△ HTC(W/M**2*K)

```
IPTS OCONEE CLAD TBVG4  10/20/83                              1. FLAWS/M**3     FO =  1.090D+19

          --------------UNADJUSTED---------------             ---ADJUSTED----
WELD      P(F/E)       95%CI     %ERR   P(INITIA)    N*V      P(F/E)    %ERR       NTRIALS


  1       2.72D-02   2.42D-03    8.90   2.75D-02   0.040    1.09D-03                 10000
  2       1.62D-02   1.33D-03    8.23   1.63D-02   0.040    6.47D-04                 20000
  3       2.63D-02   2.38D-03    9.06   2.67D-02   0.010    2.63D-04                 10000

                                                  VESSEL    2.00D-03   5.66



 DEPTHS FOR INITIAL INITIATION (MM)
                 3.2   6.3  12.7  19.0  25.4  38.1  50.8
   NUMBER        348   725   295    84    23     4     0
   PERCENT      23.5  49.0  19.9   5.7   1.6   0.3   0.0



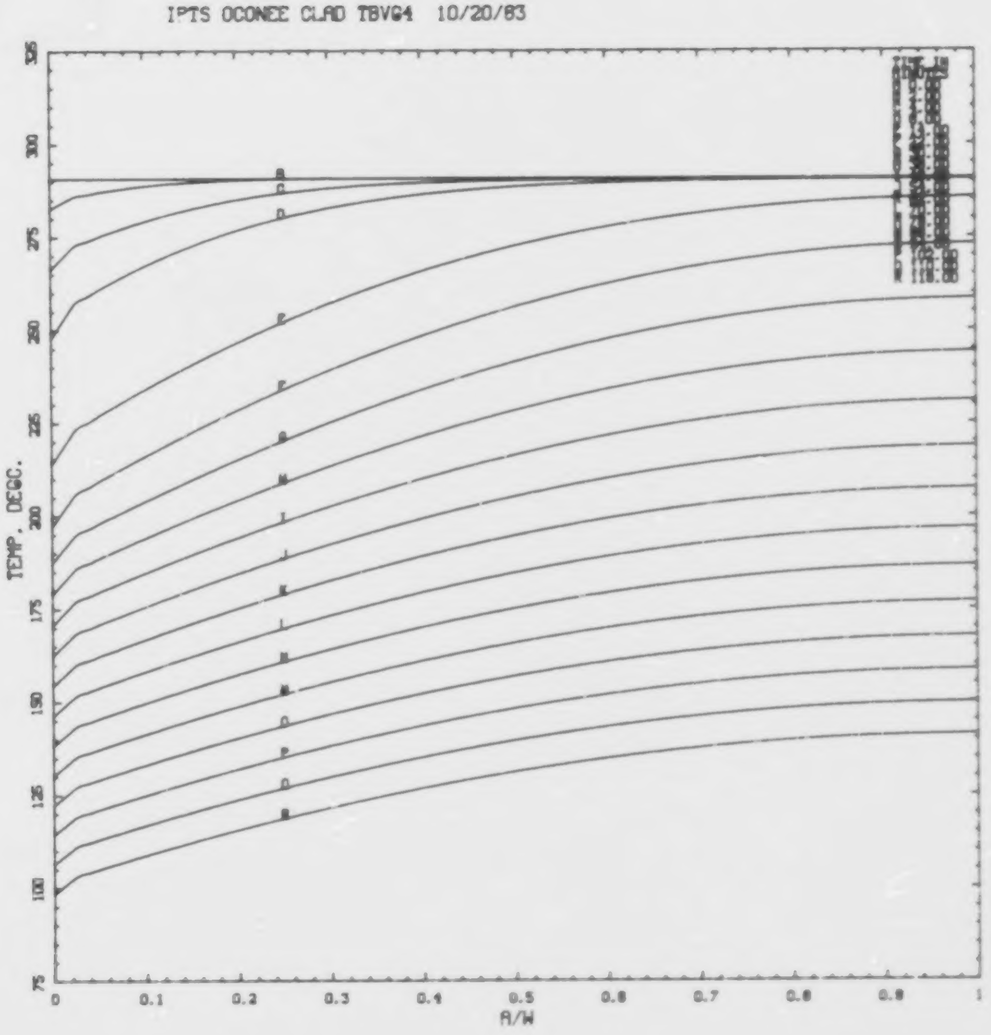
TIMES OF FAILURE(MINUTES)
                 0.0  10.0  20.0  30.0  40.0  50.0  60.0  70.0  80.0  90.0 100.0 110.0 120.0
   NUMBER              0     0     0     0     0     0     2     8    46   119   310   976
   PERCENT           0.0   0.0   0.0   0.0   0.0   0.0   0.1   0.5   3.1   8.1  21.2  66.8



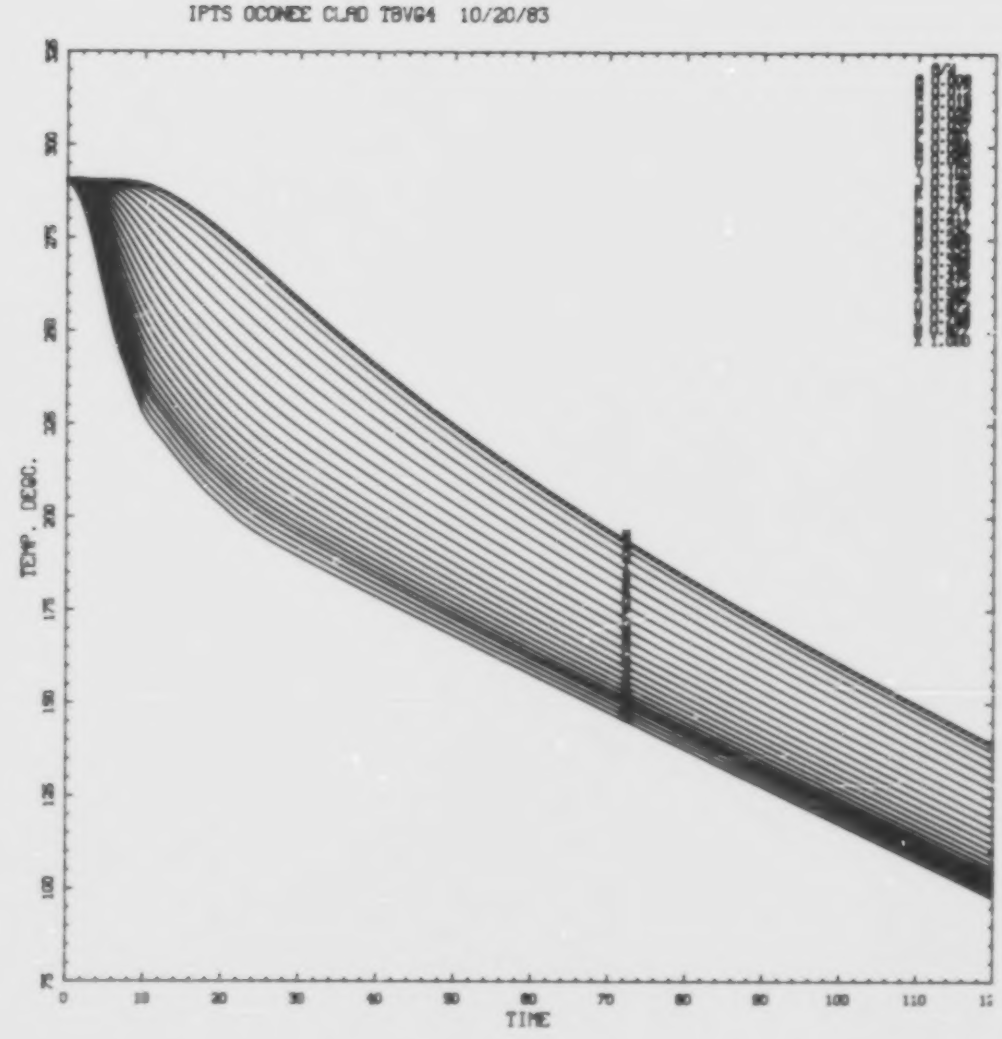
 INITIATION T-RTNDT(DEG.C)
               -55.6 -41.7 -27.8 -13.9   0.0  13.9  27.8  41.7  55.6  69.4  83.3  97.2 111.1
   NUMBER             20   113   370   580   345    51     1     0     0     0     0     0
   PERCENT           1.4   7.6  25.0  39.2  23.3   3.4   0.1   0.0   0.0   0.0   0.0   0.0



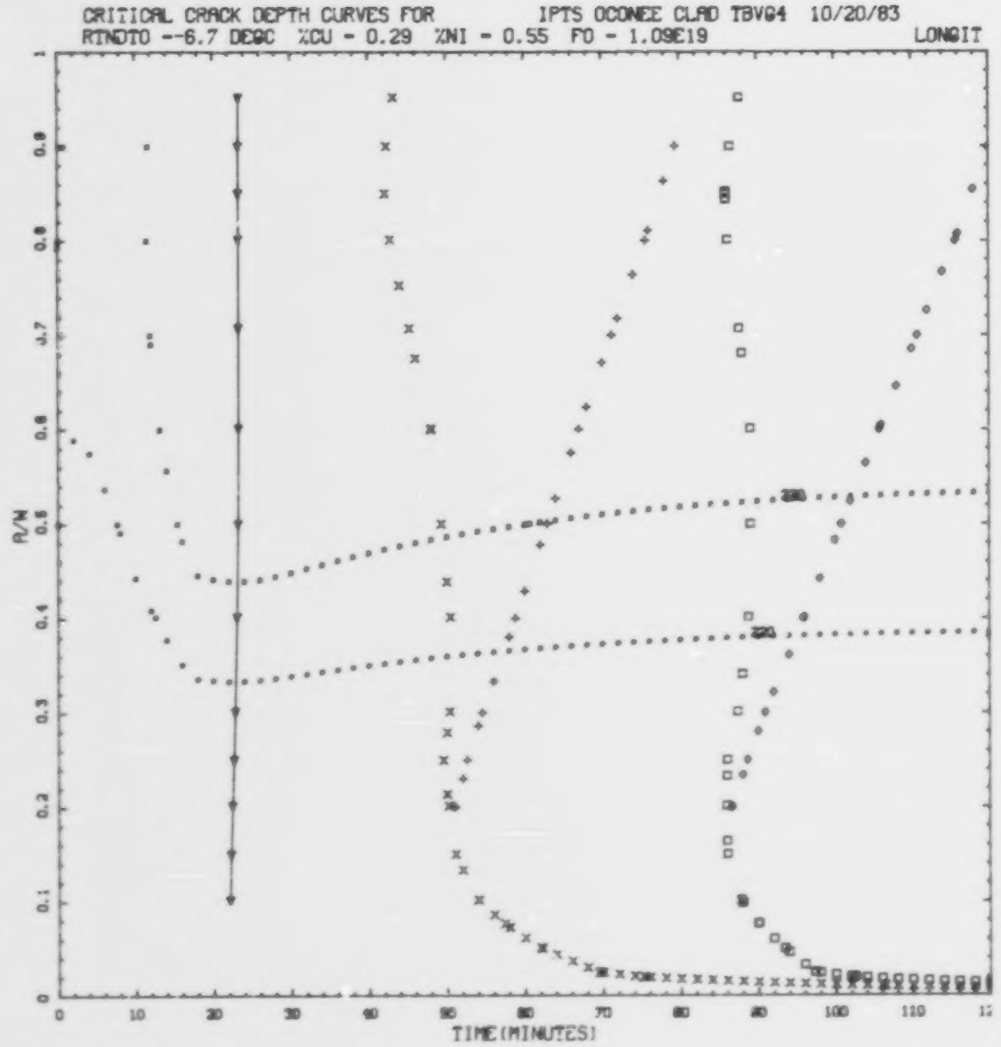
 ARREST T-RTNDT(DEG.C)
               -27.8 -13.9   0.0  13.9  27.8  41.7  55.6  69.4  83.3  97.2 111.1 125.0 138.9
   NUMBER              0     0     7    12     0     0     0     0     0     0     0     0
   PERCENT           0.0   0.0  36.8  63.2   0.0   0.0   0.0   0.0   0.0   0.0   0.0   0.0
```


IPTS OCONEE CLAD TBVG4 10/20/83
TEMP. DEGC.
75
100
125
150
175
200
225
250
275
300
325
A/W
0
0.1
0.2
0.3
0.4
0.5
0.6
0.7
0.8
0.9
1
B
C
D
E
F
G
H
I
J
K
L
M
N
O
P
Q
R

581

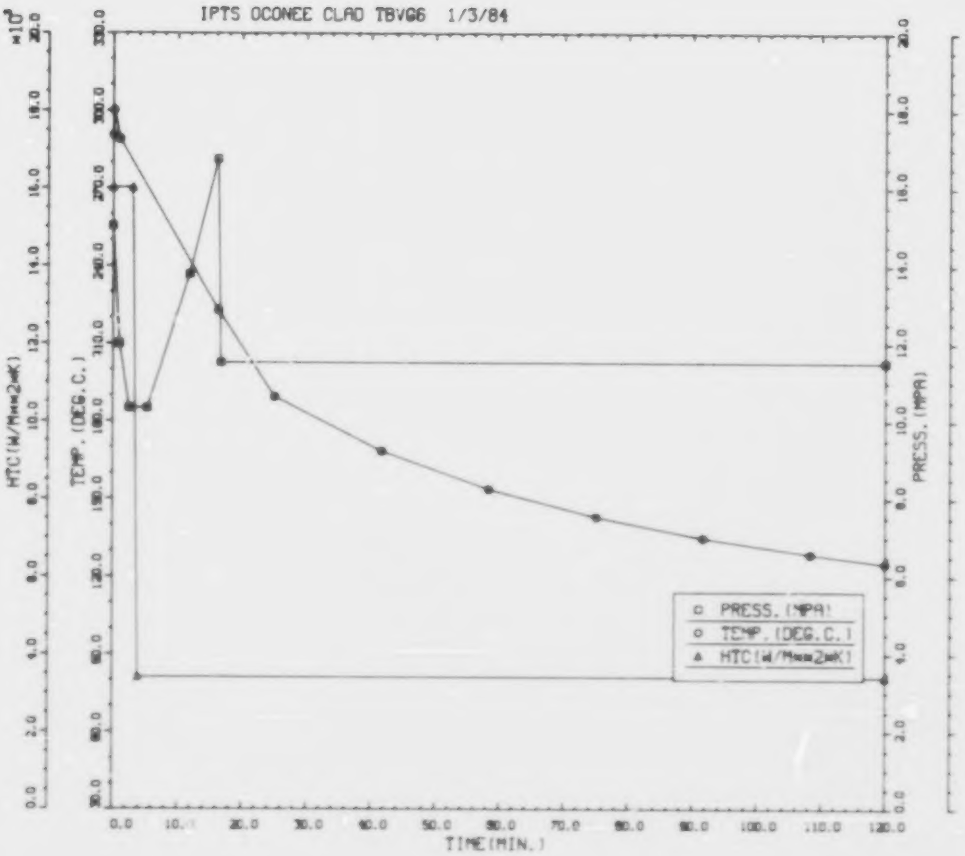

IPTS OCONEE CLAD TBVG4 10/20/83
TEMP. DEGC.
325
300
275
250
225
200
175
150
125
100
75
TIME
0
10
20
30
40
50
60
70
80
90
100
110
12

582

CRITICAL CRACK DEPTH CURVES FOR IPTS OCONEE CLAD TBV64 10/20/83
RTNDTO --6.7 DEGC %CU - 0.29 %NI - 0.55 FO - 1.09E19 LONGIT
A/W
0
0.1
0.2
0.3
0.4
0.5
0.6
0.7
0.8
0.9
1
220
220
0
10
20
30
40
50
60
70
80
90
100
110
TIME(MINUTES)

583

TRANSIENT NO. 6 (TBVG6)


IPTS OCONEE CLAD TBVG6 1/3/84
HTC(W/M**2*K) ×10³
0.0
2.0
4.0
6.0
8.0
10.0
12.0
14.0
16.0
18.0
20.0
TEMP. (DEG.C.)
30.0
60.0
90.0
120.0
150.0
180.0
210.0
240.0
270.0
300.0
330.0
PRESS. (MPA)
0.0
2.0
4.0
6.0
8.0
10.0
12.0
14.0
16.0
18.0
20.0
TIME(MIN.)
0.0
10.0
20.0
30.0
40.0
50.0
60.0
70.0
80.0
90.0
100.0
110.0
120.0
□ PRESS.(MPA)
○ TEMP.(DEG.C.)
△ HTC(W/M**2*K)

```
IPTS OCONEE CLAD TBVG6  1/3/84                          1. FLAWS/M**3     FO =  1.090D+19

          -------------UNADJUSTED-------------                ---ADJUSTED----
WELD    P(F/E)        95%CI     %ERR   P(INITIA)    N*V      P(F/E)    %ERR     NTRIALS


  1     2.23D-05   1.00D-05  44.96    4.35D-05  0.040    8.93D-07             500000
  2     1.29D-05   7.64D-06  59.10    1.88D-05  0.040    5.17D-07             500000
  3     2.58D-05   1.08D-05  41.79    3.29D-05  0.010    2.58D-07             500000

                                             VESSEL     1.67D-06  30.93



 DEPTHS FOR INITIAL INITIATION (MM)
               3.2   6.3  12.7  19.0  25.4  38.1  50.8
  NUMBER         2    43    17    10     7     2     0
  PERCENT      2.5  53.1  21.0  12.3   8.6   2.5   0.0




TIMES OF FAILURE(MINUTES)
                0.0  10.0  20.0  30.0  40.0  50.0  60.0  70.0  80.0  90.0 100.0 110.0 120.0
  NUMBER           0     0     0     0     0     0     0     3    10     7    14    18
  PERCENT        0.0   0.0   0.0   0.0   0.0   0.0   0.0   5.8  19.2  13.5  26.9  34.6

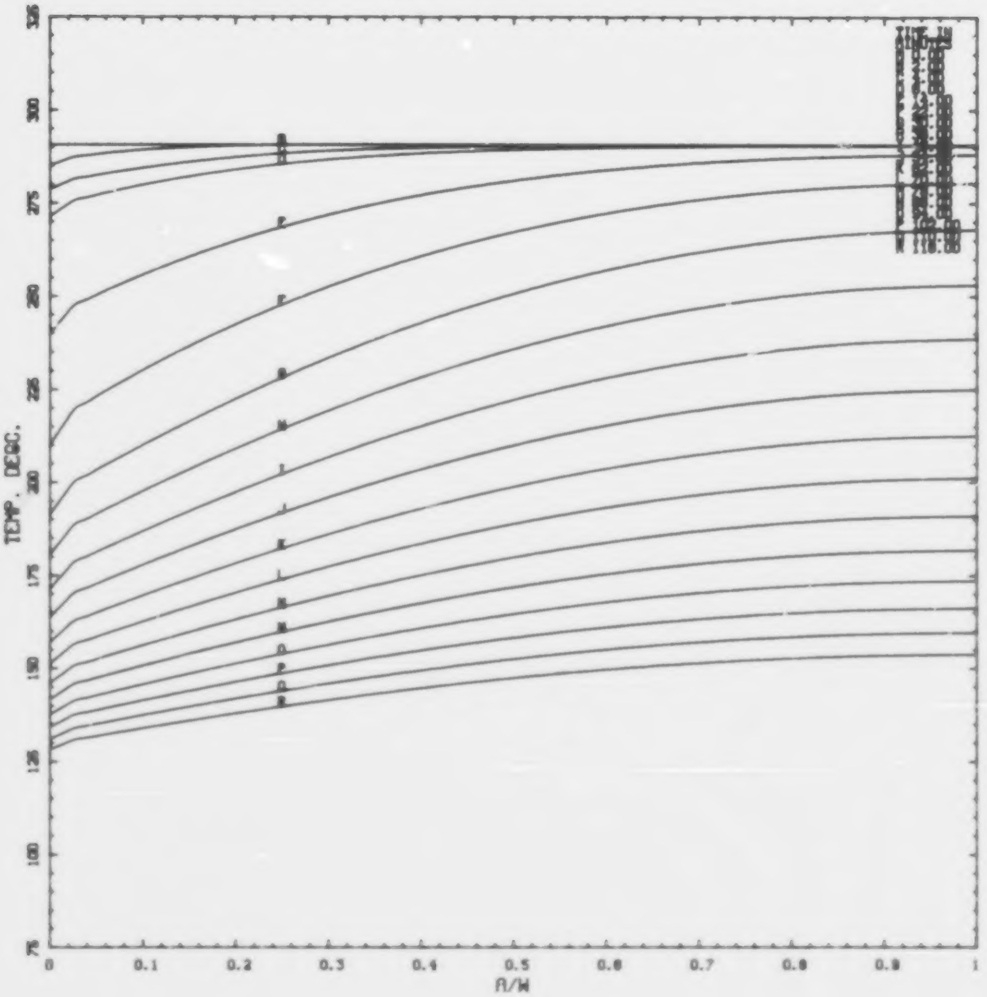


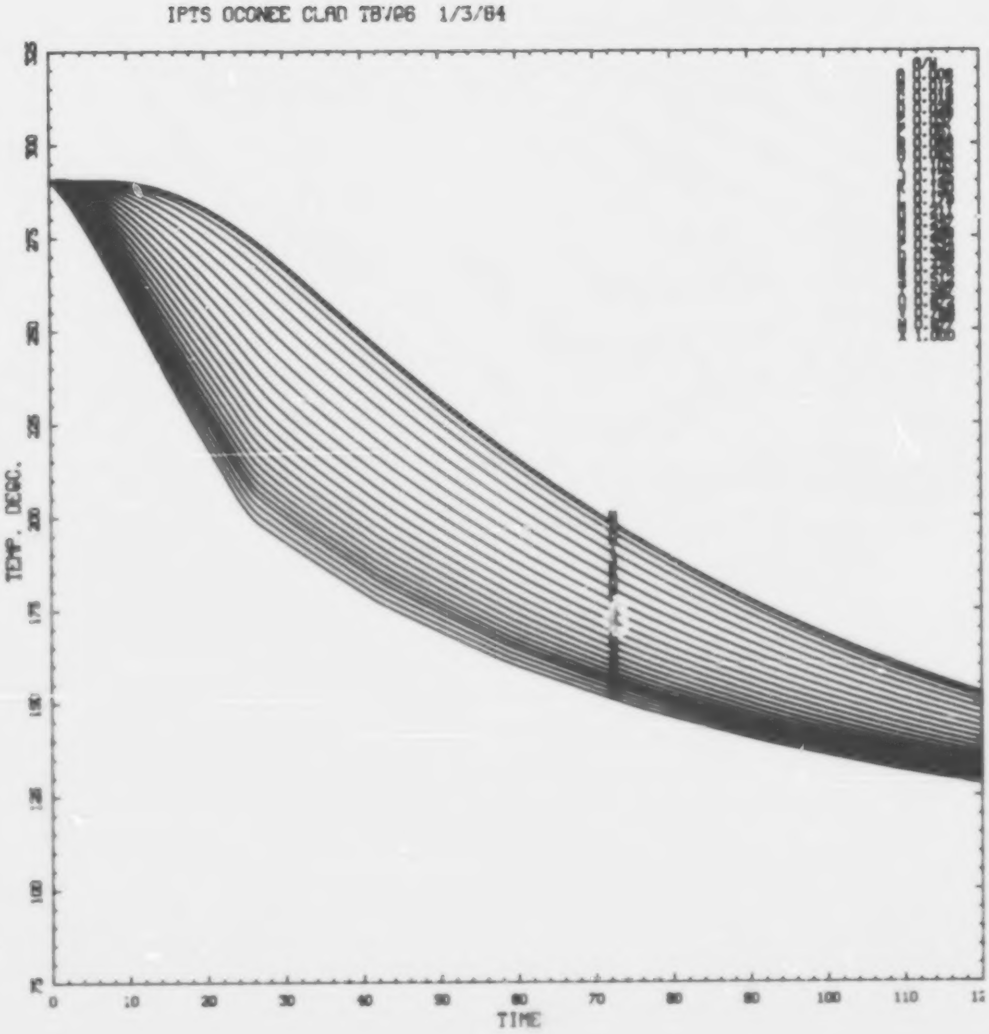

 INITIATION T-RTNDT(DEG.C)
              -55.6 -41.7 -27.8 -13.9   0.0  13.9  27.8  41.7  55.6  69.4  83.3  97.2 111.1
  NUMBER           0     0    15    37    29     1     4     1     0     0     0     0
  PERCENT        0.0   0.0  17.2  42.5  33.3   1.1   4.6   1.1   0.0   0.0   0.0   0.0




 ARREST T-RTNDT(DEG.C)
              -27.8 -13.9   0.0  13.9  27.8  41.7  55.6  69.4  83.3  97.2 111.1 125.0 138.9
  NUMBER           2    14    13     0     0     0     0     4     2     0     0     0
  PERCENT        5.7  40.0  37.1   0.0   0.0   0.0   0.0  11.4   5.7   0.0   0.0   0.0
```

IPTS OCONEE CLAD T8/06 1/3/84

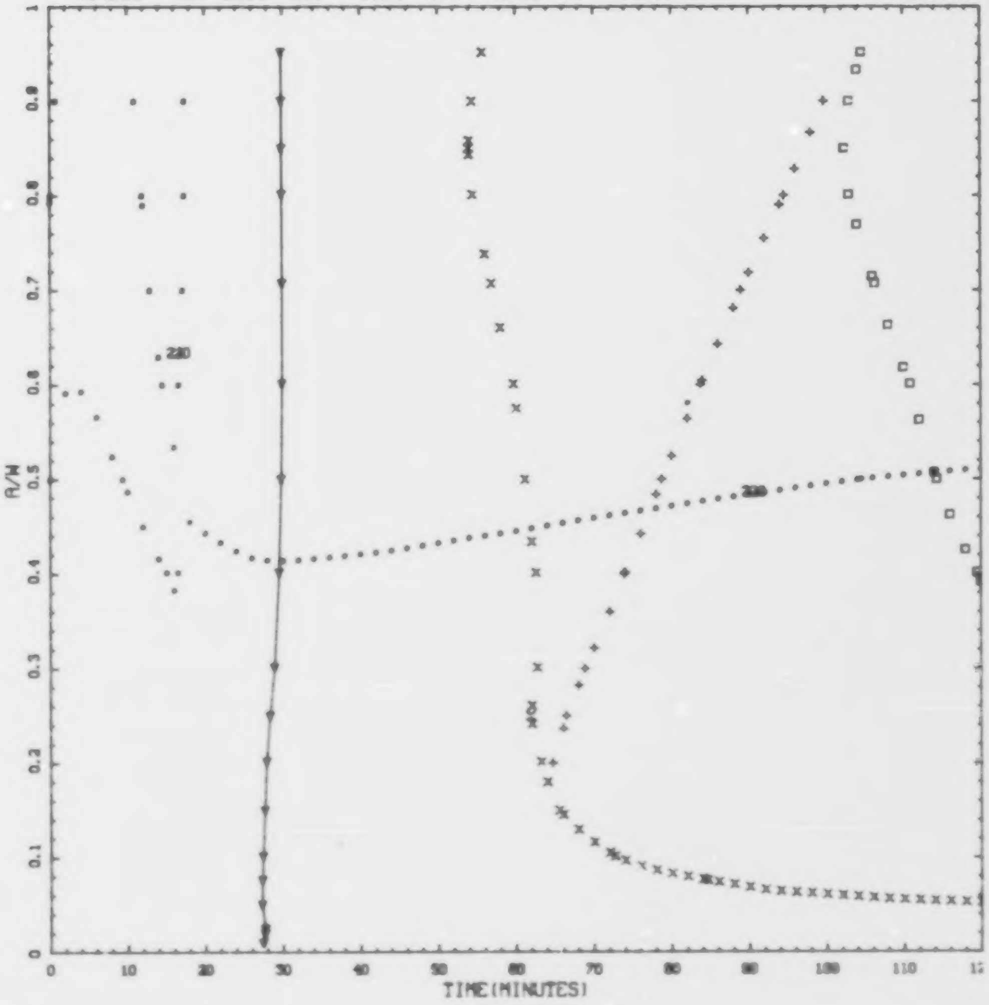




IPTS OCONEE CLAD TB/06 1/3/84

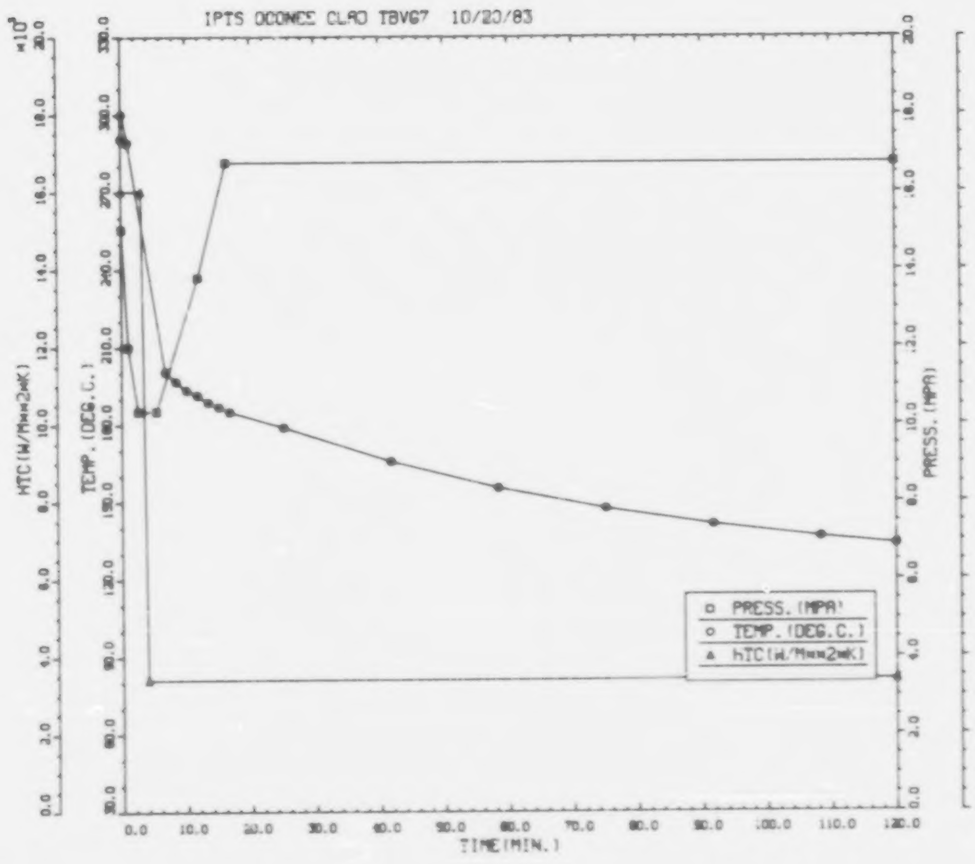
588

CRITICAL CRACK DEPTH CURVES FOR IPTS OCONEE CLAD TBVG6 1/3/84
RTNDTO --6.7 DEGC %CU - 0.29 %NI - 0.55 FO - 1.09E19 LONGIT



TRANSIENT NO. 7 (TBVG7)

540


IPTS OCONEE CLAD TBVG7 10/20/83
HTC(W/M**2*K)
*10^3
0.0
2.0
4.0
6.0
8.0
10.0
12.0
14.0
16.0
18.0
20.0
TEMP.(DEG.C.)
30.0
60.0
90.0
120.0
150.0
180.0
210.0
240.0
270.0
300.0
330.0
PRESS.(MPA)
0.0
2.0
4.0
6.0
8.0
10.0
12.0
14.0
16.0
18.0
20.0
TIME(MIN.)
0.0
10.0
20.0
30.0
40.0
50.0
60.0
70.0
80.0
90.0
100.0
110.0
120.0
□ PRESS.(MPA)
○ TEMP.(DEG.C.)
▲ HTC(W/M**2*K)

```
IPTS OCONEE CLAD TBVG7  10/20/83                         1. FLAWS/M**3     FO =  1.090D+19

        -------------UNADJUSTED---------------            ---ADJUSTED----
WELD    P(F/E)       95%CI    %ERR   P(INITIA)   N*V      P(F/E)    %ERR      NTRIALS


 1      8.11D-05   1.91D-05  23.59   9.28D-05  0.040   3.24D-06                500000
 2      2.94D-05   1.15D-05  39.20   3.17D-05  0.040   1.17D-06                500000
 3      5.46D-05   1.95D-05  23.10   8.69D-05  0.010   8.46D-07                500000

                                           VESSEL     5.26D-06  17.37



 DEPTHS FOR INITIAL INITIATION (MM)
                 3.2   6.3  12.7  19.0  25.4  38.1  50.8
   NUMBER         6    82    43    31    15     3     0
   PERCENT      3.3  45.6  23.9  17.2   8.3   1.7   0.0



TIMES OF FAILURE(MINUTES)
                 0.0  10.0  20.0  30.0  40.0  50.0  60.0  70.0  80.0  90.0 100.0 110.0 120.0
   NUMBER           0     0     0     0     1     4     8    18    23    30    34    48
   PERCENT        0.0   0.0   0.0   0.0   0.6   2.4   4.8  10.8  13.9  18.1  20.5  28.9

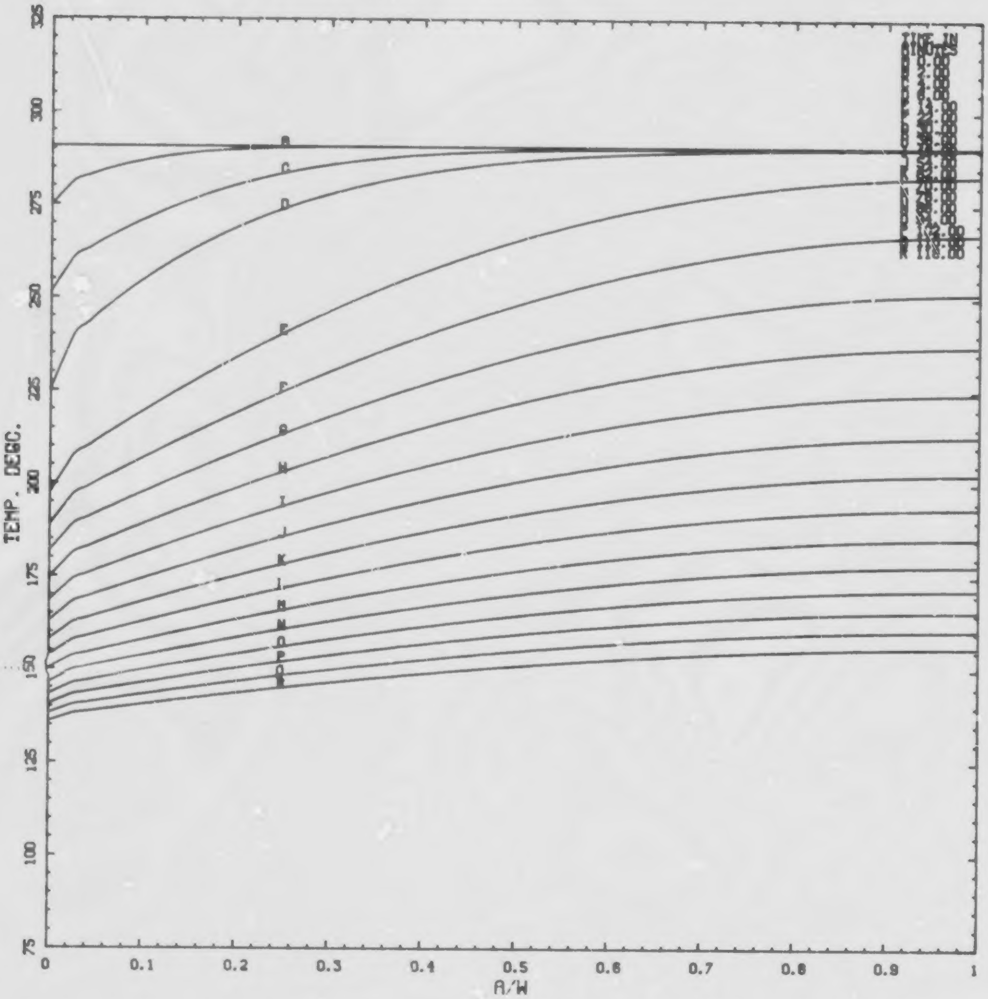


 INITIATION T-RTNDT(DEG.C)
              -55.6 -41.7 -27.8 -13.9   0.0  13.9  27.8  41.7  55.6  69.4  83.3  97.2 111.1
   NUMBER           0     1    16    56    84    22     1     0     0     0     0     0
   PERCENT        0.0   0.6   8.9  31.1  46.7  12.2   0.6   0.0   0.0   0.0   0.0   0.0



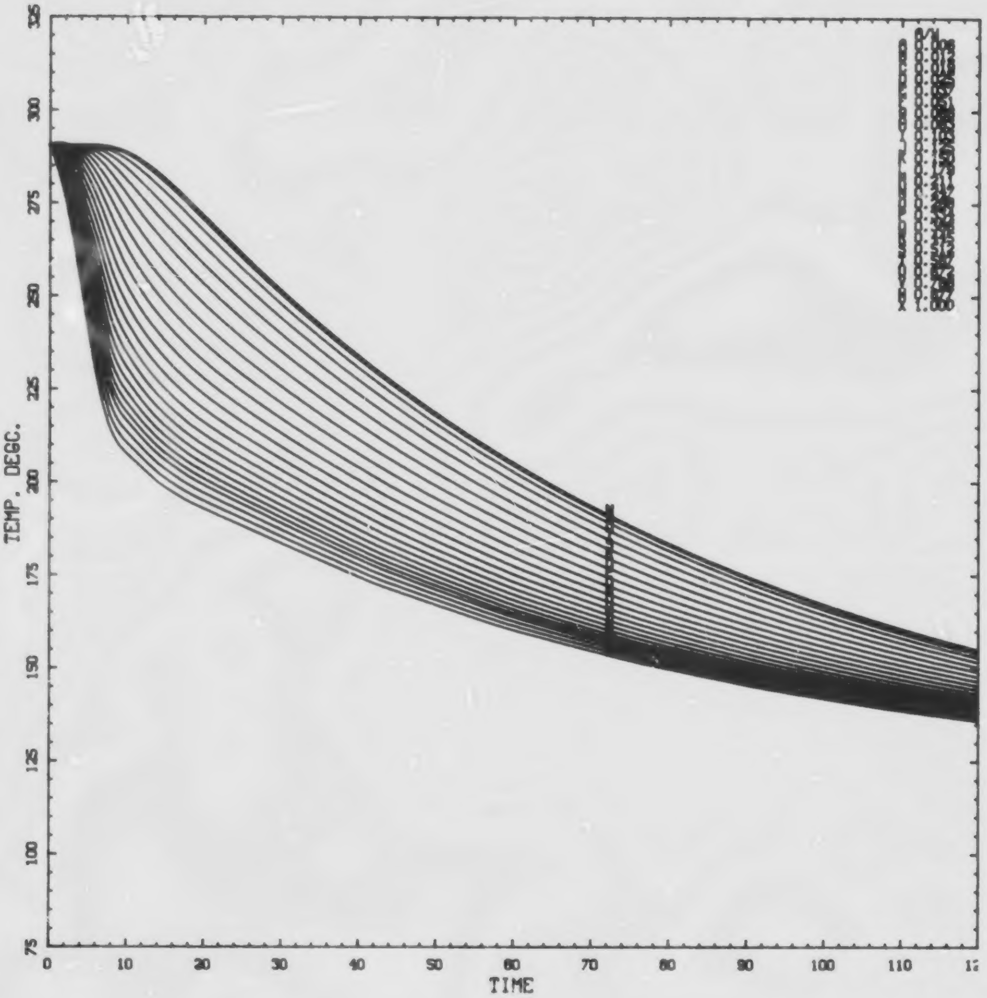
 ARREST T-RTNDT(DEG.C)
              -27.8 -13.9   0.0  13.9  27.8  41.7  55.6  69.4  83.3  97.2 111.1 125.0 138.9
   NUMBER           0     2    12     0     0     0     0     0     0     0     0     0
   PERCENT        0.0  14.3  85.7   0.0   0.0   0.0   0.0   0.0   0.0   0.0   0.0   0.0
```

IPTS OCONEE CLAD TBVG7 10/20/83

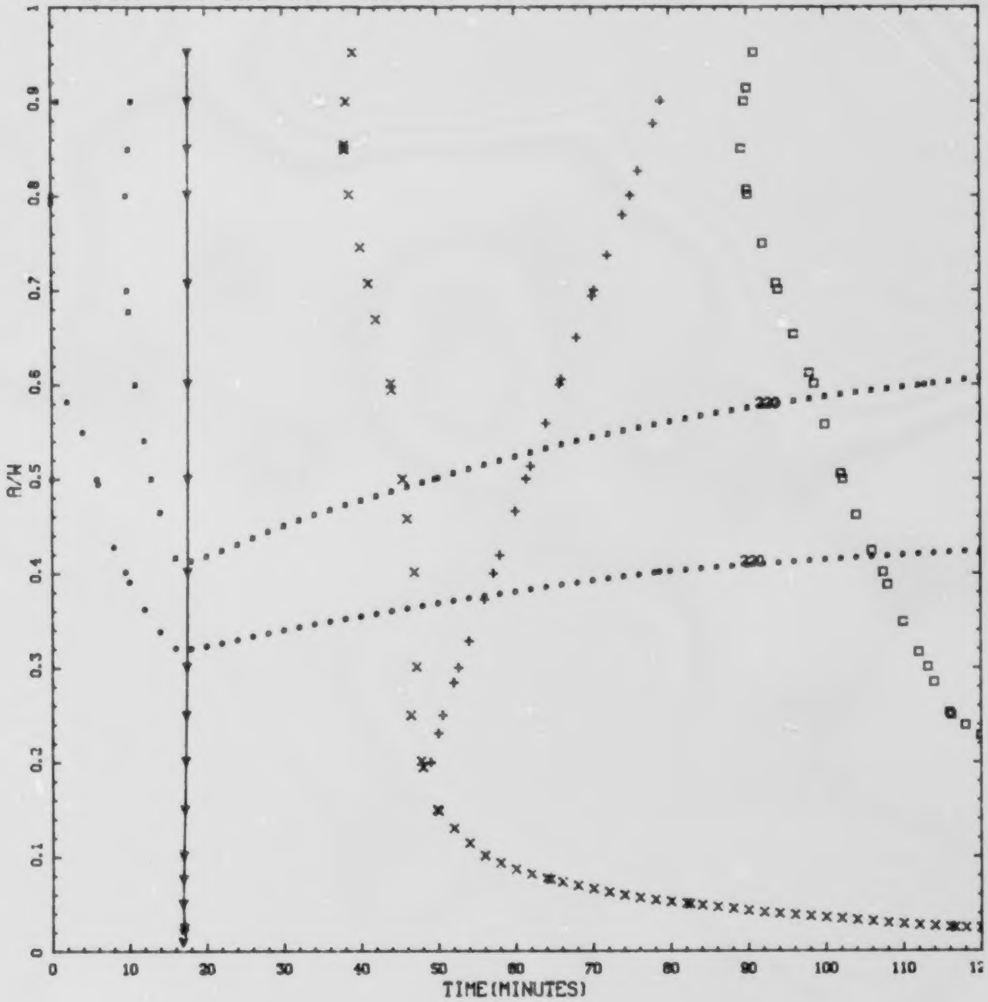


593

IPTS OCONEE CLAD TBVG7 10/20/83

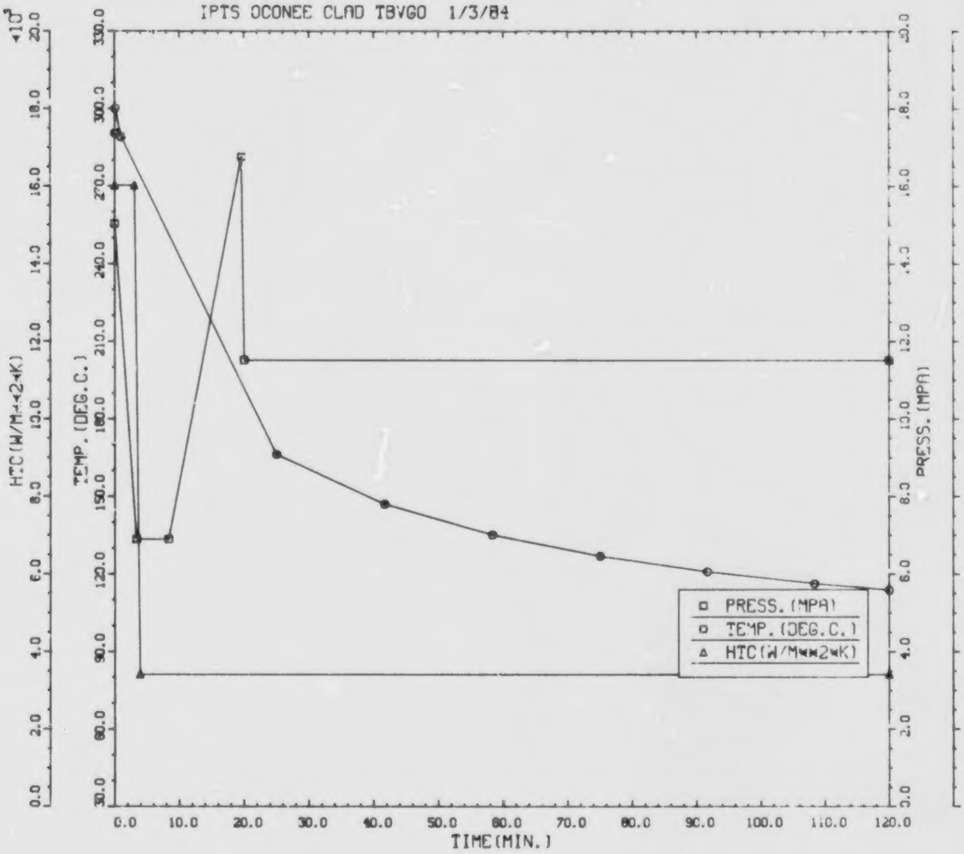


CRITICAL CRACK DEPTH CURVES FOR PTS OCONEE CLAD TB/G7 10/20/83
RTNDTO --6.7 DEGC %CU - 0.29 %NI - 0.55 FO - 1.09E19 LONGIT



TRANSIENT NO. 10 [TBVG10(TBVG0)]

546


IPTS OCONEE CLAD TBVGO 1/3/84
HTC(W/M**2*K) ×10³
TEMP.(DEG.C.)
PRESS.(MPA)
TIME(MIN.)
0.0
2.0
4.0
6.0
8.0
10.0
12.0
14.0
16.0
18.0
20.0
30.0
60.0
90.0
120.0
150.0
180.0
210.0
240.0
270.0
300.0
330.0
10.0
20.0
30.0
40.0
50.0
60.0
70.0
80.0
90.0
100.0
110.0
120.0
□ PRESS.(MPA)
○ TEMP.(DEG.C.)
△ HTC(W/M**2*K)

```
IPTS OCONEE CLAD TBVG0  1/3/84                             1. FLAWS/M**3     F0 =  1.090D+19

        --------------UNADJUSTED---------------            ---ADJUSTED----
WELD    P(F/E)       95%CI    %ERR   P(INITIA)    N*V      P(F/E)    %ERR     NTRIALS


 1      1.46D-04   2.56D-05  17.60   2.06D-04  0.040    5.83D-06             500000
 2      6.46D-05   1.71D-05  26.43   9.63D-05  0.040    2.58D-06             500000
 3      1.41D-04   2.52D-05  17.89   1.96D-04  0.010    1.41D-06             500000

                                            VESSEL      9.82D-06  12.81



 DEPTHS FOR INITIAL INITIATION (MM)
                 3.2   6.3  12.7  19.0  25.4  38.1  50.8
   NUMBER         24   265    64    43    24     4     0
   PERCENT       5.7  62.5  15.1  10.1   5.7   0.9   0.0



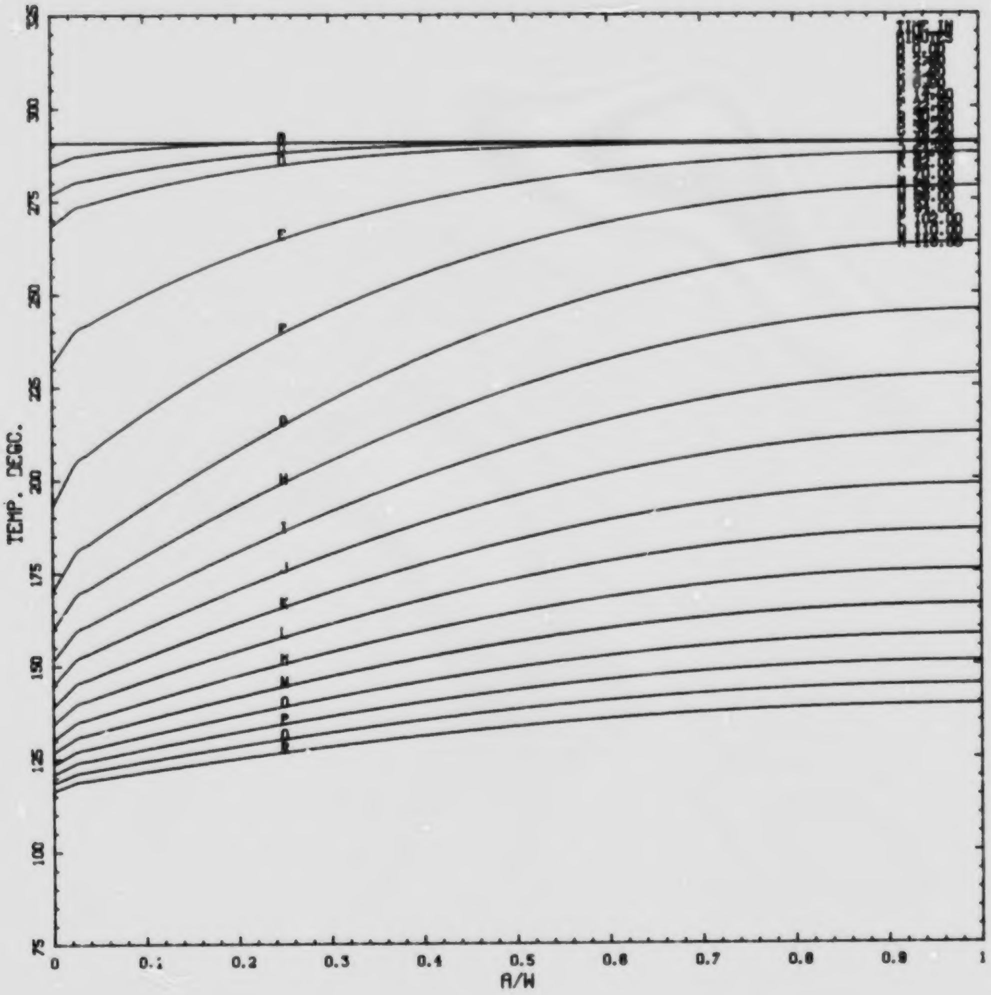

TIMES OF FAILURE(MINUTES)
                 0.0  10.0  20.0  30.0  40.0  50.0  60.0  70.0  80.0  90.0 100.0 110.0 120.0
   NUMBER            0     0     0     0     0     5    23    71    63    61    42    34
   PERCENT         0.0   0.0   0.0   0.0   0.0   1.7   7.7  23.7  21.1  20.4  14.0  11.4




 INITIATION T-RTNDT(DEG.C)
               -55.6 -41.7 -27.8 -13.9   0.0  13.9  27.8  41.7  55.6  69.4  83.3  97.2 111.1
   NUMBER            0     0    30   180   172    63   106    32     0     0     0     0
   PERCENT         0.0   0.0   5.1  30.9  29.5  10.8  18.2   5.5   0.0   0.0   0.0   0.0



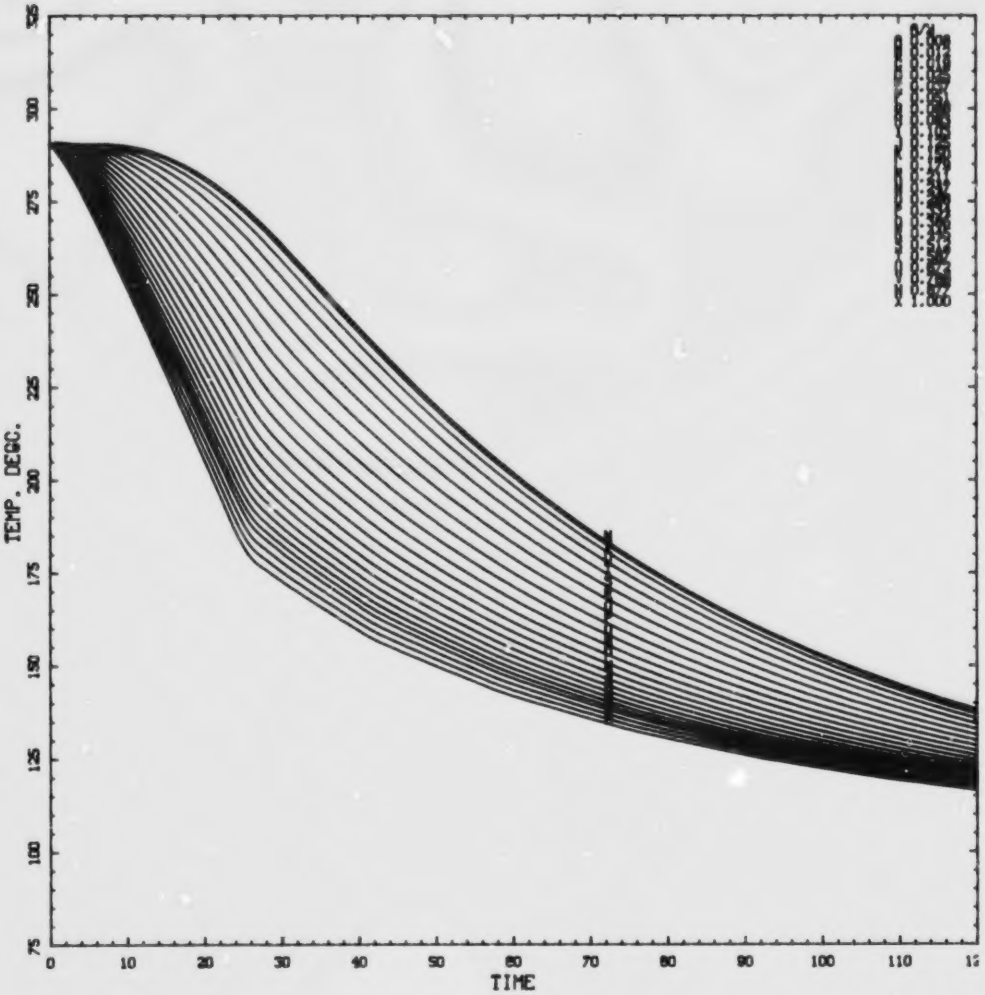

 ARREST T-RTNDT(DEG.C)
               -27.8 -13.9   0.0  13.9  27.8  41.7  55.6  69.4  83.3  97.2 111.1 125.0 138.9
   NUMBER            1    50    45     6     0     0     4    84    94     0     0     0
   PERCENT         0.4  17.6  15.8   2.1   0.0   0.0   1.4  29.6  33.1   0.0   0.0   0.0
```



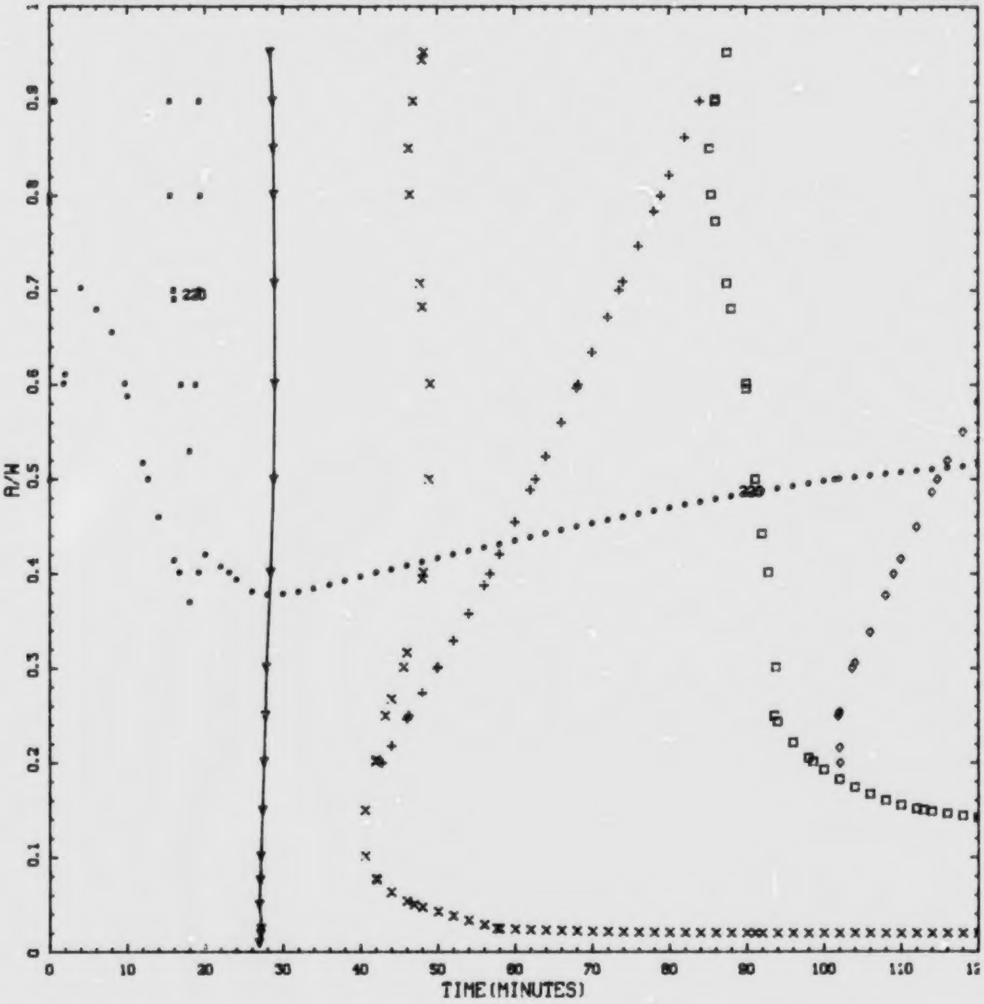
IPTS OCONEE CLAD TBVGO 1/3/84

599

IPTS OCONEE CLAD TBVGO 1/3/84



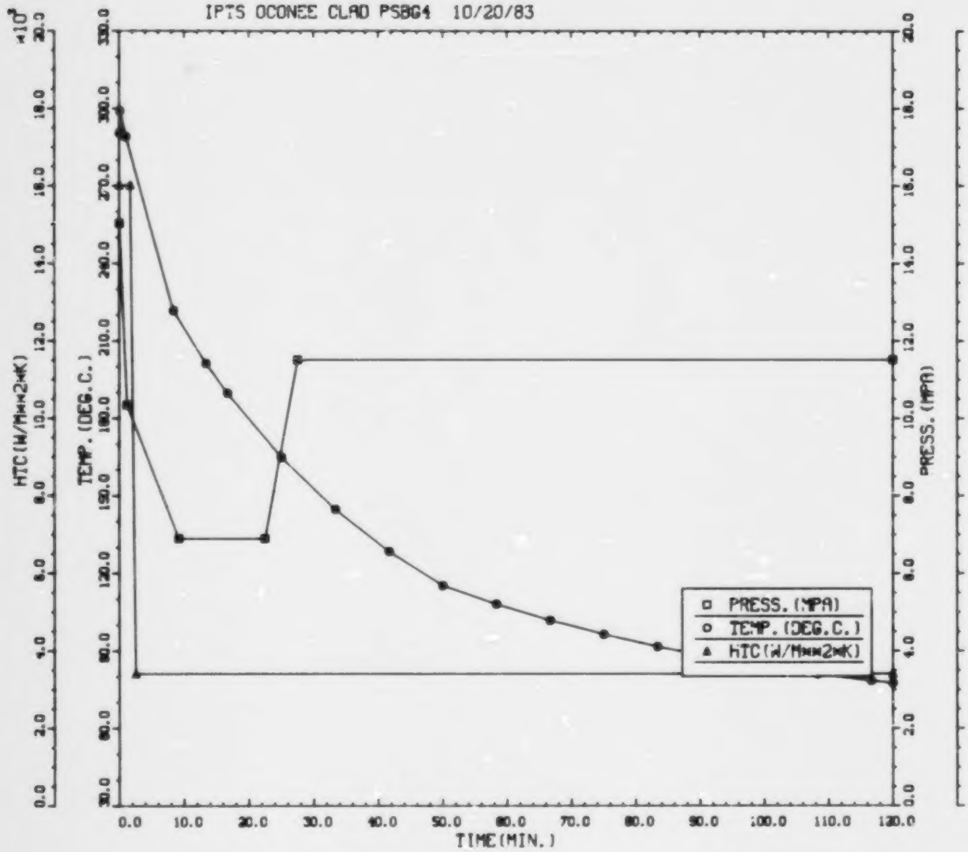
600


CRITICAL CRACK DEPTH CURVES FOR IPTS OCONEE CLAD TBVGO 1/3/84
RTNDTO --6.7 DEGC %CU - 0.29 %NI - 0.55 FC - 1.09E19 LONGIT

601

TRANSIENT NO. 14 (PSBG4)

BEST COPY AVAILABLE


IPTS OCONEE CLAD PSBG4 10/20/83
HTC(W/M**2*K)
*10³
0.0
2.0
4.0
6.0
8.0
10.0
12.0
14.0
16.0
18.0
20.0
TEMP.(DEG.C.)
30.0
60.0
90.0
120.0
150.0
180.0
210.0
240.0
270.0
300.0
330.0
PRESS.(MPA)
0.0
2.0
4.0
6.0
8.0
10.0
12.0
14.0
16.0
18.0
20.0
TIME(MIN.)
0.0
10.0
20.0
30.0
40.0
50.0
60.0
70.0
80.0
90.0
100.0
110.0
120.0
□ PRESS.(MPA)
○ TEMP.(DEG.C.)
▲ HTC(W/M**2*K)

603

```
IPTS OCONEE CLAD PSBG4  10/20/83                        1. FLAWS/M**3     FO =  1.090D+19

          --------------UNADJUSTED---------------           ---ADJUSTED----
WELD      P(F/E)       95%CI     %ERR   P(INITIA)   N*V      P(F/E)     %ERR     NTRIALS


  1       1.59D-02   1.32D-03    8.30   1.72D-02  0.040    6.38D-04                20000
  2       1.13D-02   9.12D-04    8.09   1.25D-02  0.040    4.51D-04                30000
  3       1.65D-02   1.34D-03    8.16   1.73D-02  0.010    1.65D-04                20000

                                              VESSEL       1.25D-03   5.24


 DEPTHS FOR INITIAL INITIATION (MM)
                3.2   6.3  12.7  19.0  25.4  38.1  50.8
   NUMBER       376  1049   255    94    33     5     0
   PERCENT     20.8  57.9  14.1   5.2   1.8   0.3   0.0



TIMES OF FAILURE(MINUTES)
                  0.0  10.0  20.0  30.0  40.0  50.0  60.0  70.0  80.0  90.0 100.0 110.0 120.0
   NUMBER           0     0     0     0     0    14   141   370   412   373   235   135
   PERCENT        0.0   0.0   0.0   0.0   0.0   0.8   8.4  22.0  24.5  22.2  14.0   8.0

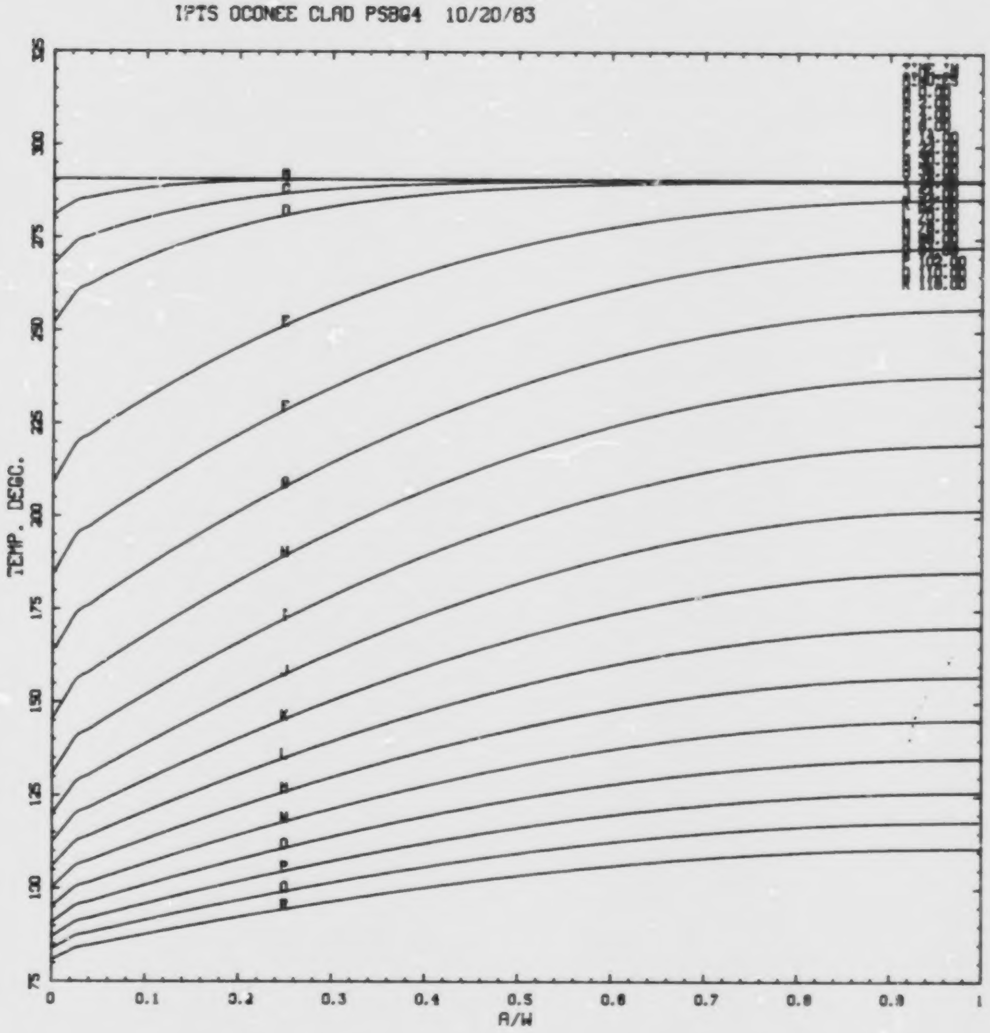


 INITIATION T-RTNDT(DEG.C)
               -55.6 -41.7 -27.8 -13.9   0.0  13.9  27.8  41.7  55.6  69.4  83.3  97.2 111.1
   NUMBER           0    28   424   783   496   172   507   175     3     0     0     0
   PERCENT        0.0   1.1  16.4  30.3  19.2   6.6  19.6   6.8   0.1   0.0   0.0   0.0

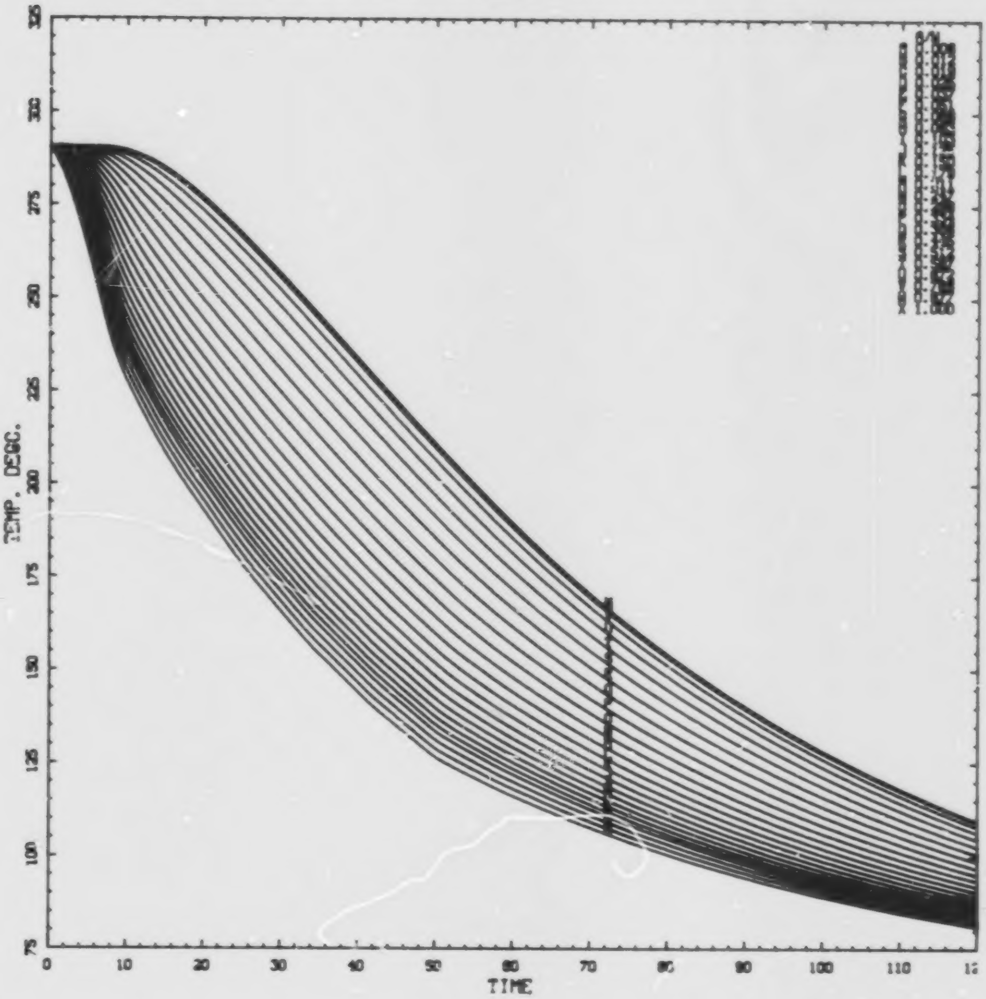


 ARREST T-RTNDT(DEG.C)
               -27.8 -13.9   0.0  13.9  27.8  41.7  55.6  69.4  83.3  97.2 111.1 125.0 138.9
   NUMBER           5    31    50    20     1     0    14   262   523     2     0     0
   PERCENT        0.6   3.4   5.5   2.2   0.1   0.0   1.5  28.9  57.6   0.2   0.0   0.0
```

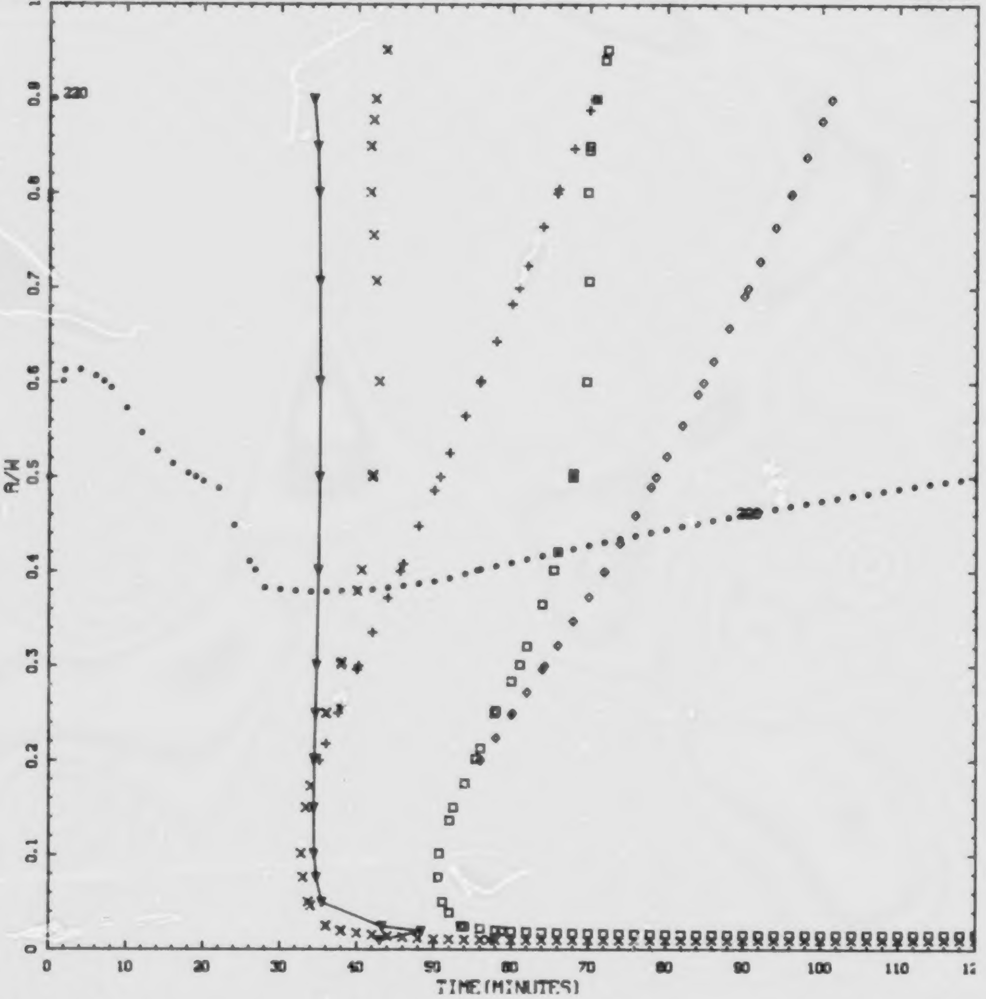
604


IPTS OCONEE CLAD PS804 10/20/83
TEMP. DEGC.
A/W
75
100
125
150
175
200
225
250
275
300
325
0
0.1
0.2
0.3
0.4
0.5
0.6
0.7
0.8
0.9
1
B
C
D
E
F
G
H
I
J
K
L
M
N
O
P
Q
R

IPTS OCONEE CLAD PSBG4 10/20/83

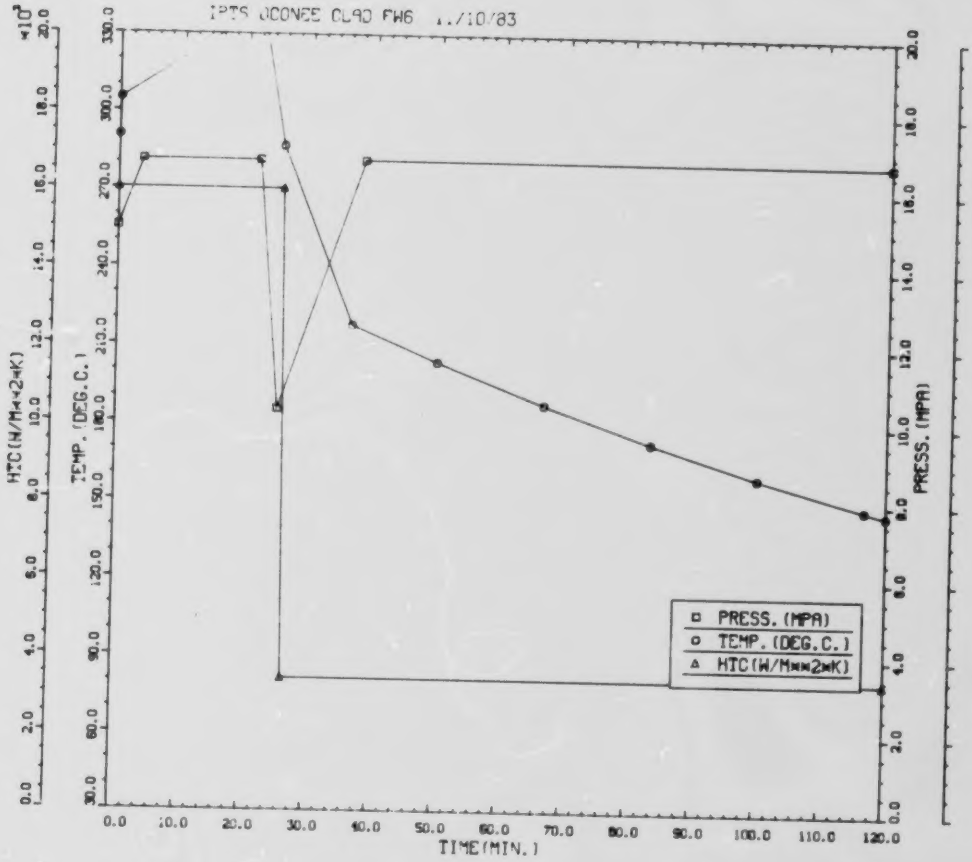


CRITICAL CRACK DEPTH CURVES FOR IPTS OCONEE CLAD PSBG4 10/20/83
RTNDTO --6.7 DEGC %CU - 0.29 %NI - 0.55 FO - 1.09E19 LONGIT



607

TRANSIENT NO. 21 [FW6(OVRFD6)]


IPTS OCONEE CL9D FW6 1/10/83
HTC(W/M**2*K) *10³
0.0
2.0
4.0
6.0
8.0
10.0
12.0
14.0
16.0
18.0
20.0
TEMP.(DEG.C.)
30.0
60.0
90.0
120.0
150.0
180.0
210.0
240.0
270.0
300.0
330.0
PRESS.(MPA)
0.0
2.0
4.0
6.0
8.0
10.0
12.0
14.0
16.0
18.0
20.0
TIME(MIN.)
0.0
10.0
20.0
30.0
40.0
50.0
60.0
70.0
80.0
90.0
100.0
110.0
120.0
□ PRESS.(MPA)
○ TEMP.(DEG.C.)
△ HTC(W/M**2*K)

```
IPTS OCONEE CLAD FW6  11/10/83                         1. FLAWS/M**3     F0 =  1.090D+19

       -------------UNADJUSTED---------------          ---ADJUSTED----
WELD    P(F/E)      95%CI    %ERR   P(INITIA)   N*V     P(F/E)    %ERR    NTRIALS


 1      1.53D-05   8.30D-06  54.36   1.64D-05  0.040  6.11D-07            500000
 2      3.52D-06   3.99D-06 113.16   3.52D-06  0.040  1.41D-07            500000
 3      1.76D-05   8.92D-06  50.61   1.76D-05  0.010  1.76D-07            500000

                                            VESSEL     9.28D-07  40.84



 DEPTHS FOR INITIAL INITIATION (MM)
              3.2    6.3   12.7   19.0   25.4   38.1   50.8
   NUMBER       1     11     13      4      3      0      0
   PERCENT    3.1   34.4   40.6   12.5    9.4    0.0    0.0



TIMES OF FAILURE(MINUTES)
                 0.0  10.0  20.0  30.0  40.0  50.0  60.0  70.0  80.0  90.0 100.0 110.0 120.0
   NUMBER           0     0     0     0     0     0     0     0     0     0     5    26
   PERCENT        0.0   0.0   0.0   0.0   0.0   0.0   0.0   0.0   0.0   0.0  16.1  83.9

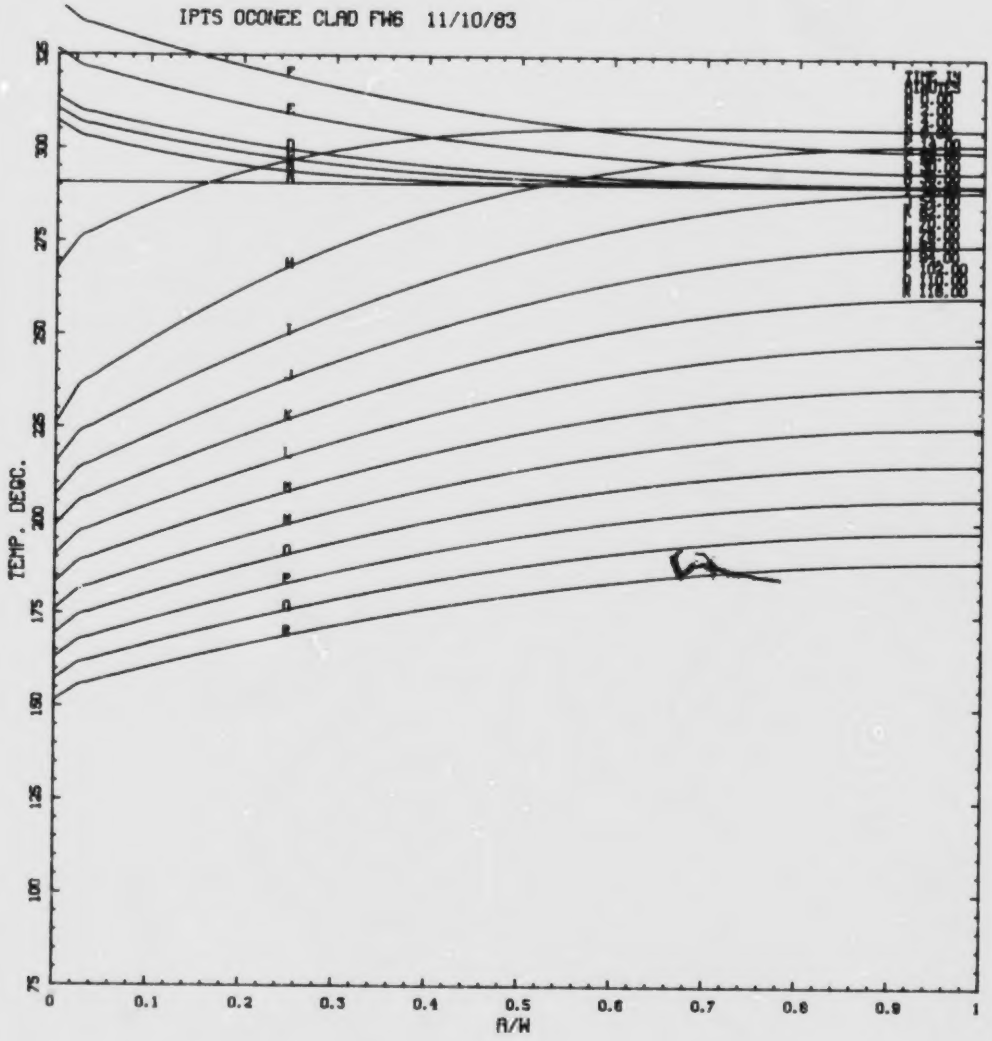


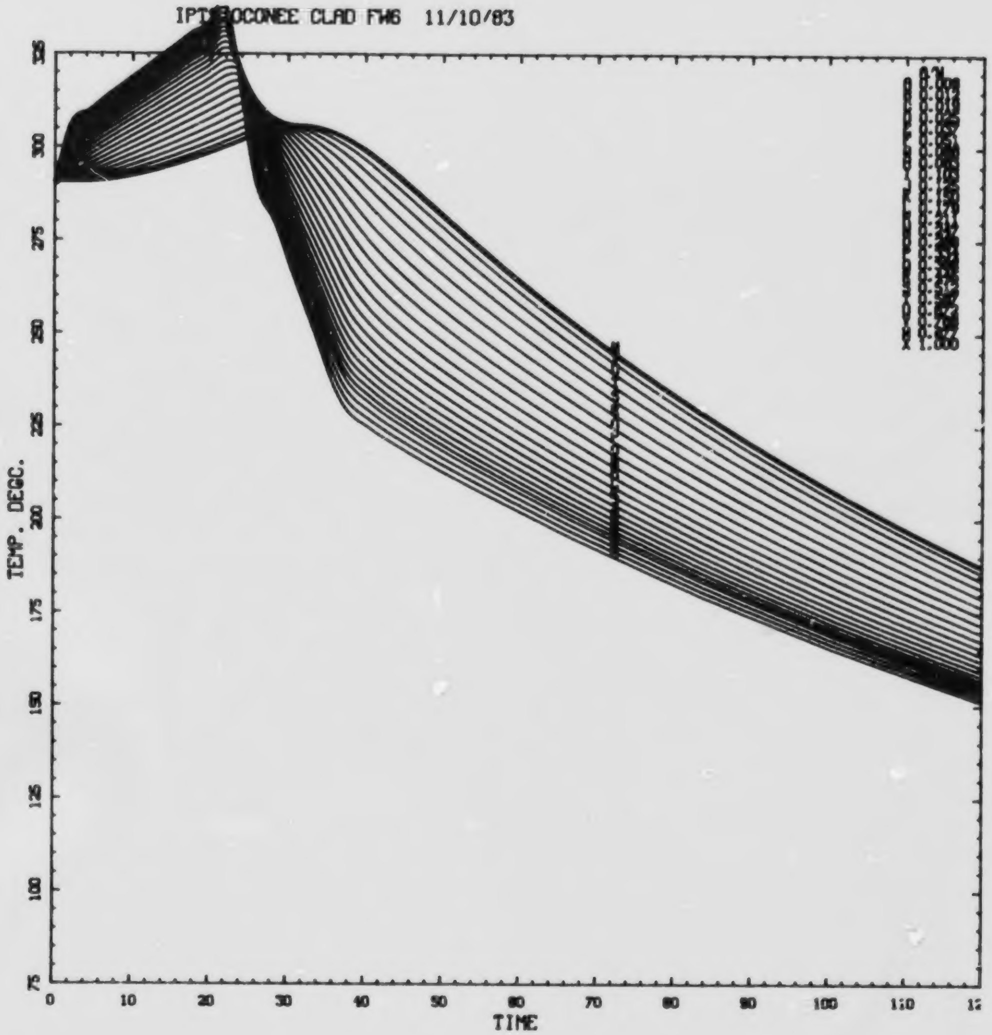
 INITIATION T-RTNDT(DEG.C)
              -55.6 -41.7 -27.8 -13.9   0.0  13.9  27.8  41.7  55.6  69.4  83.3  97.2 111.1
   NUMBER           0     0     0     4    23     5     0     0     0     0     0     0
   PERCENT        0.0   0.0   0.0  12.5  71.9  15.6   0.0   0.0   0.0   0.0   0.0   0.0

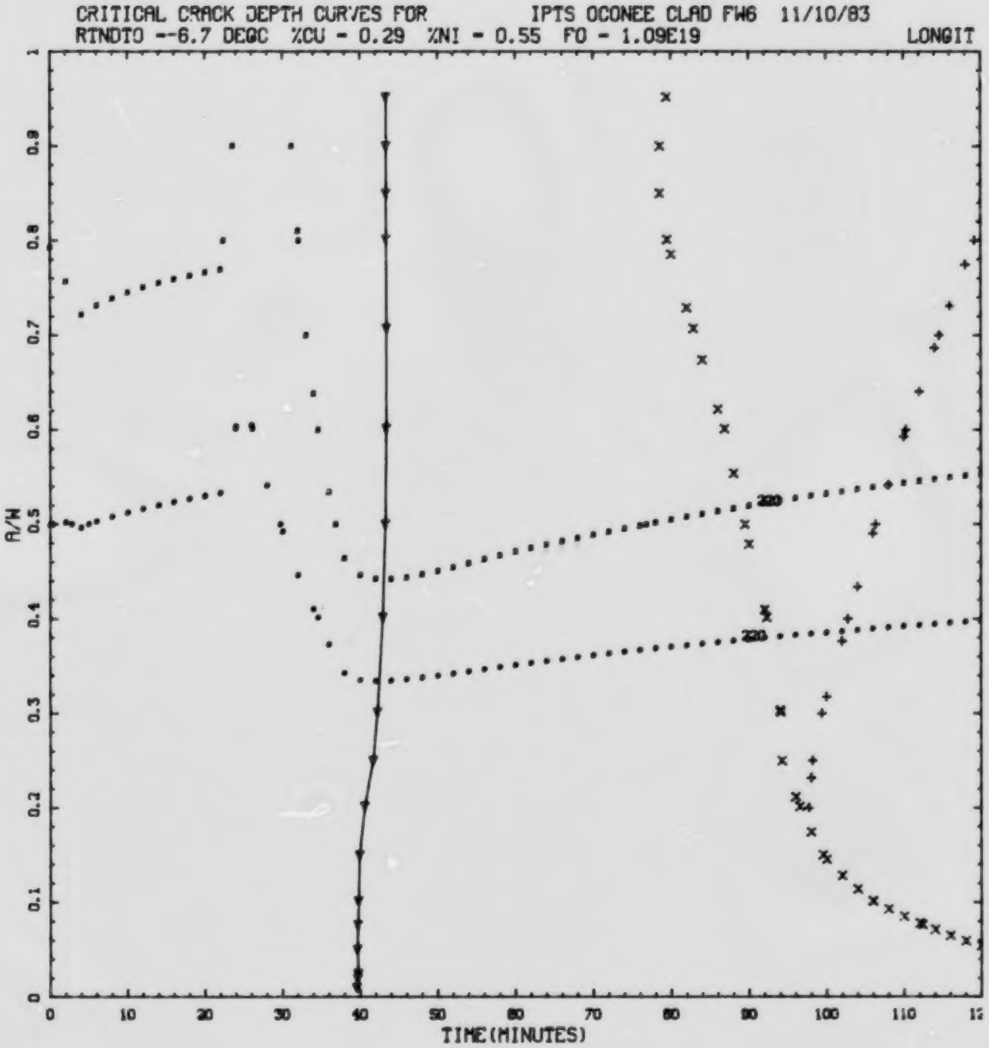


 ARREST T-RTNDT(DEG.C)
              -27.8 -13.9   0.0  13.9  27.8  41.7  55.6  69.4  83.3  97.2 111.1 125.0 138.9
   NUMBER           0     0     1     0     0     0     0     0     0     0     0     0
   PERCENT        0.0   0.0 100.0   0.0   0.0   0.0   0.0   0.0   0.0   0.0   0.0   0.0
```

IPTS OCONEE CLAD FW6 11/10/83

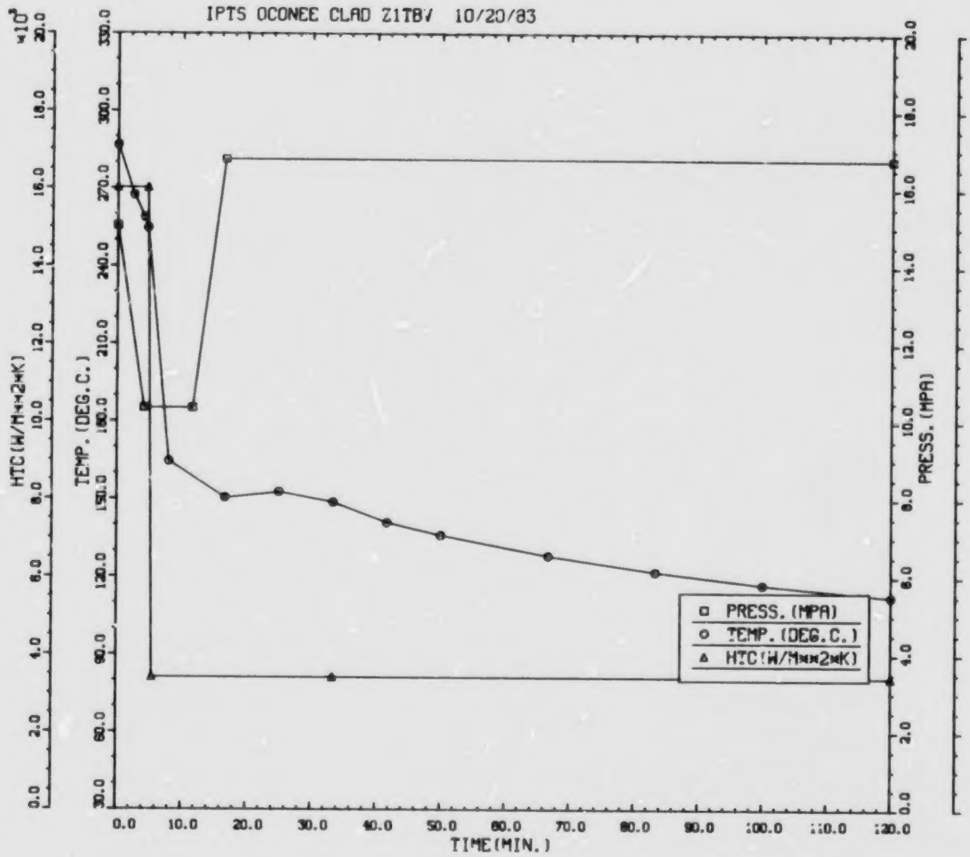



IPT OCONEE CLAD FW6 11/10/83
TEMP. DEGC.
325
300
275
250
225
200
175
150
125
100
75
TIME
0
10
20
30
40
50
60
70
80
90
100
110
12
X 1.000



CRITICAL CRACK DEPTH CURVES FOR IPTS OCONEE CLAD FW6 11/10/83
RTNDTO --6.7 DEGC %CU - 0.29 %NI - 0.55 FO - 1.09E19 LONGIT

TRANSIENT NO. 24 [Z1TBV(ZP1TBV)]


IPTS OCONEE CLAD Z1TBV 10/20/83
×10³
HTC(W/M**2*K)
0.0
2.0
4.0
6.0
8.0
10.0
12.0
14.0
16.0
18.0
20.0
TEMP.(DEG.C.)
30.0
60.0
90.0
120.0
150.0
180.0
210.0
240.0
270.0
300.0
330.0
PRESS.(MPA)
0.0
2.0
4.0
6.0
8.0
10.0
12.0
14.0
16.0
18.0
20.0
TIME(MIN.)
0.0
10.0
20.0
30.0
40.0
50.0
60.0
70.0
80.0
90.0
100.0
110.0
120.0
□ PRESS.(MPA)
○ TEMP.(DEG.C.)
△ HTC(W/M**2*K)

```
IPTS OCONEE CLAD Z1TBV  10/20/83                         1. FLAWS/M**3     F0 =  1.090D+19

          -------------UNADJUSTED--------------          ---ADJUSTED----
WELD    P(F/E)      95%CI     %ERR   P(INITIA)   N*V    P(F/E)    %ERR    NTRIALS


  1    1.51D-03   1.46D-04   9.64   1.60D-03  0.040   6.05D-05             160000
  2    6.98D-04   6.90D-05   9.89   7.44D-04  0.040   2.79D-05             330000
  3    1.65D-03   1.63D-04   9.87   1.74D-03  0.010   1.65D-05             140000

                                         VESSEL     1.05D-04    6.35



 DEPTHS FOR INITIAL INITIATION (MM)
                 3.2   6.3  12.7  19.0  25.4  38.1  50.8
  NUMBER          83   702   283   133    51    15     2
  PERCENT        6.5  55.3  22.3  10.5   4.0   1.2   0.2



TIMES OF FAILURE(MINUTES)
                 0.0  10.0  20.0  30.0  40.0  50.0  60.0  70.0  80.0  90.0 100.0 110.0 120.0
  NUMBER              0     9     2    31   116   112   165   135   154   180   126   167
  PERCENT           0.0   0.8   0.2   2.6   9.7   9.4  13.8  11.3  12.9  15.0  10.5  14.0

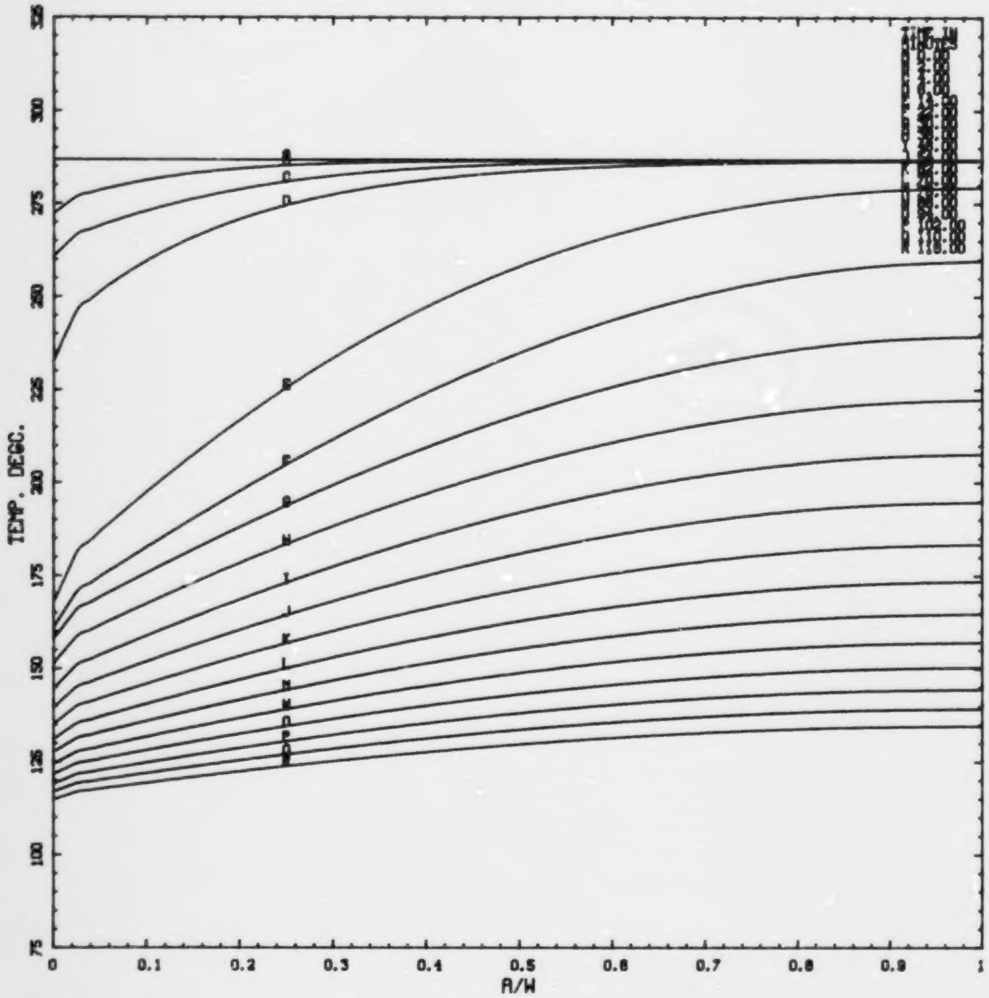


 INITIATION T-RTNDT(DEG.C)
               -55.6 -41.7 -27.8 -13.9   0.0  13.9  27.8  41.7  55.6  69.4  83.3  97.2 111.1
  NUMBER              0     2   130   483   514   129    17     2     0     0     0     0
  PERCENT           0.0   0.2  10.2  37.8  40.3  10.1   1.3   0.2   0.0   0.0   0.0   0.0



 ARREST T-RTNDT(DEG.C)
               -27.8 -13.9   0.0  13.9  27.8  41.7  55.6  69.4  83.3  97.2 111.1 125.0 138.9
  NUMBER              2    17    45    10     1     0     0     1     4     0     0     0
  PERCENT           2.5  21.3  56.3  12.5   1.3   0.0   0.0   1.3   5.0   0.0   0.0   0.0
```

IPTS OCONEE CLAD Z1TBV 10/20/83



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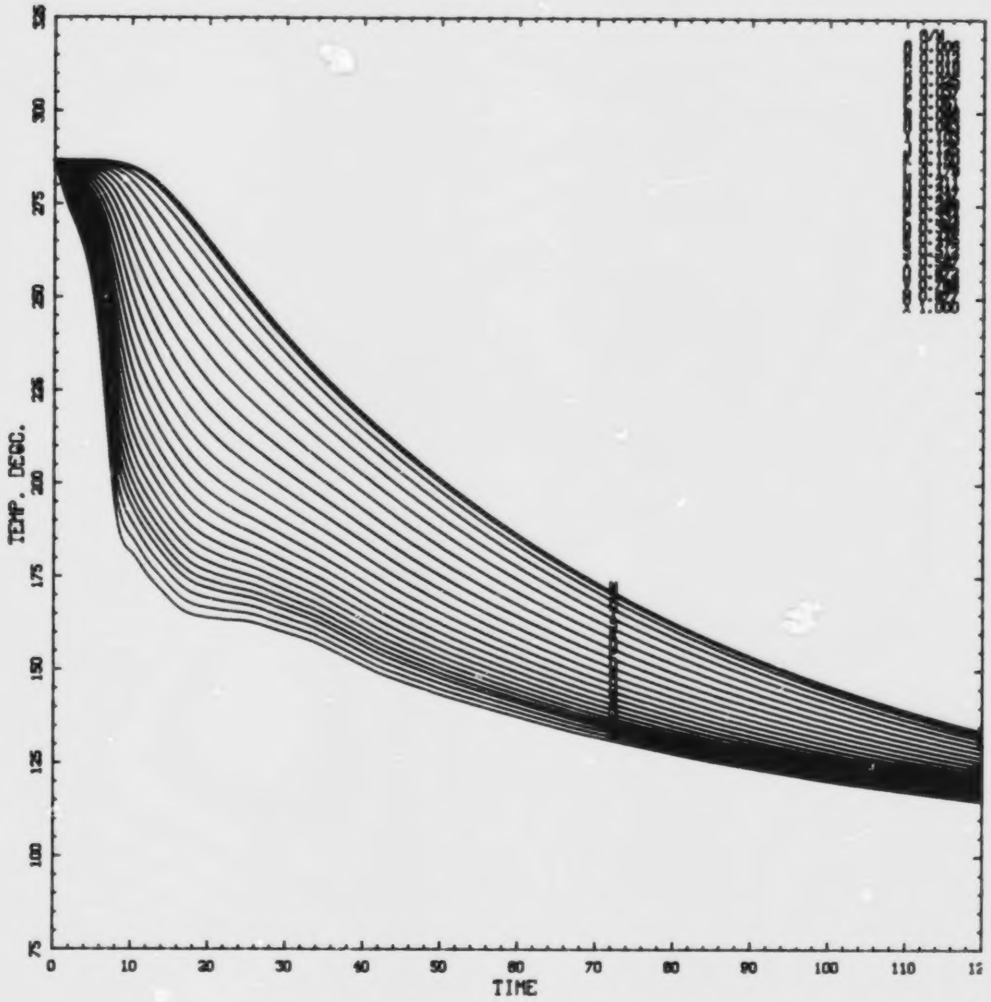
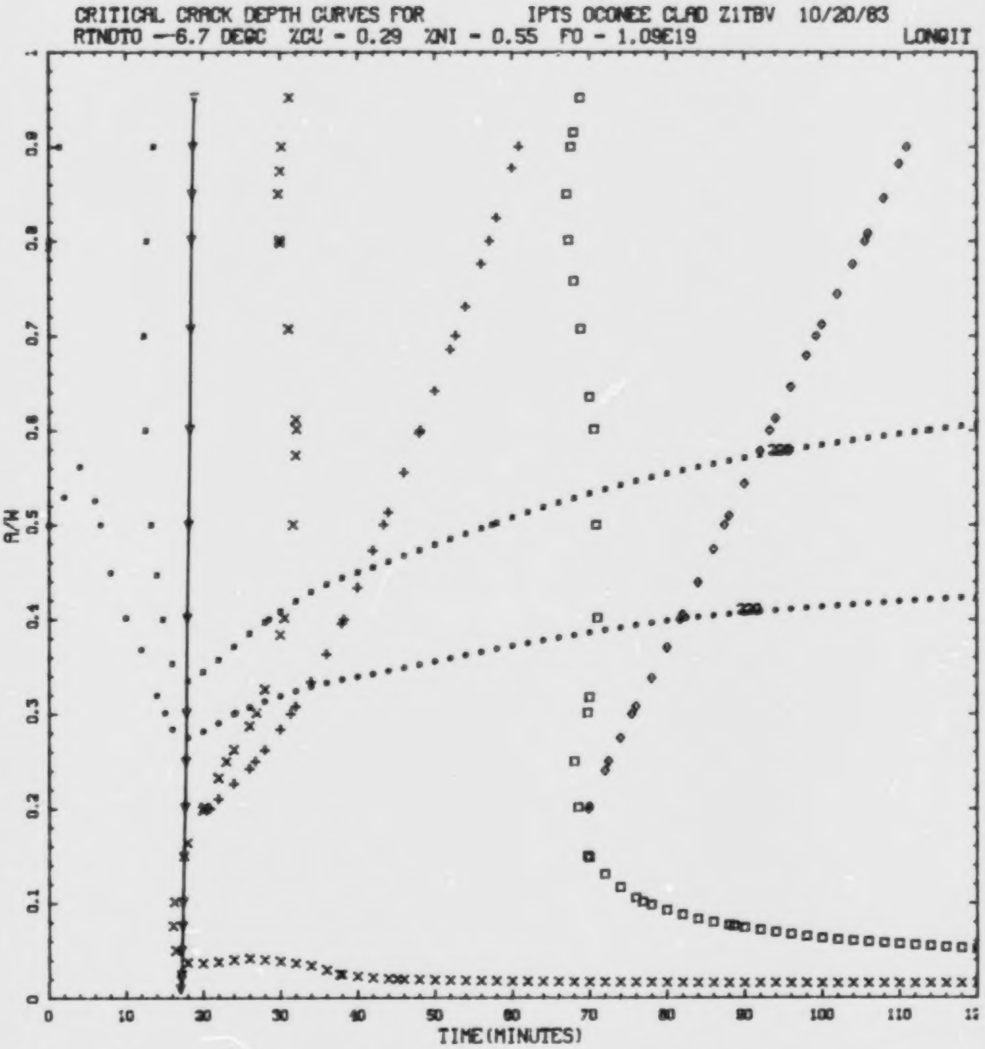
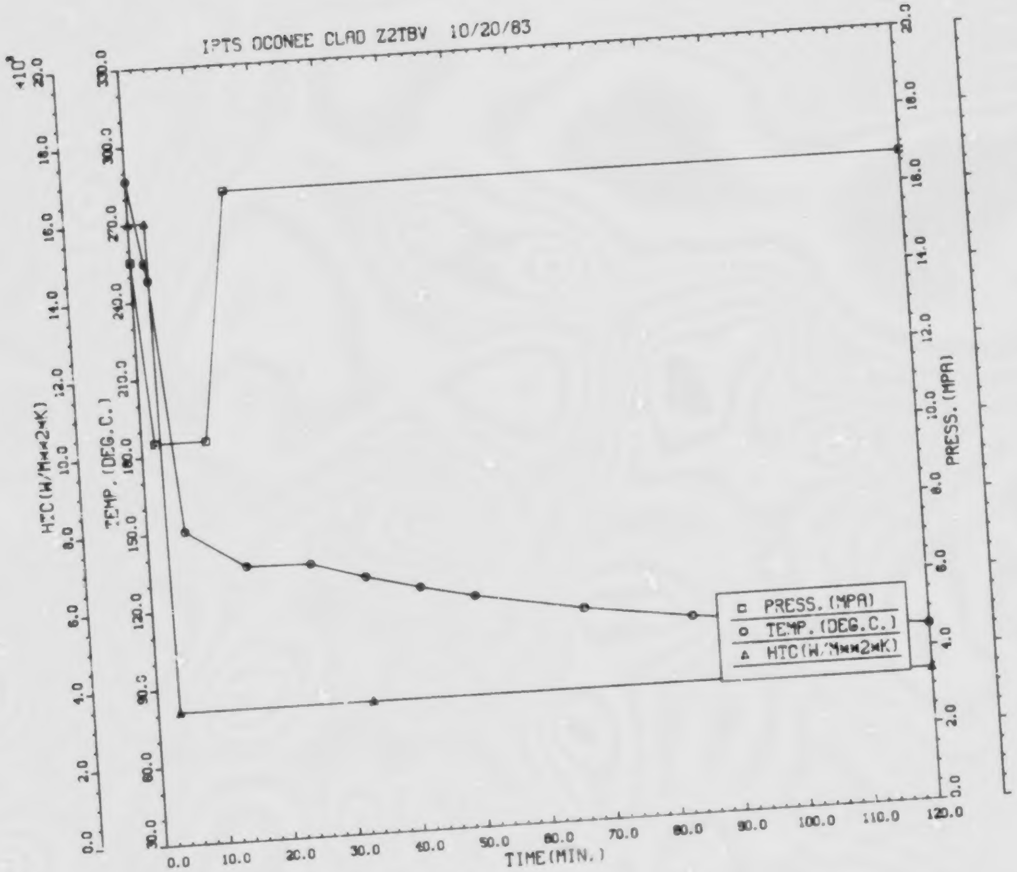
IPTS OCONEE CLAD Z1TBV 10/20/83




CRITICAL CRACK DEPTH CURVES FOR IPTS OCONEE CLAD Z1TBV 10/20/83
RTNDTO --6.7 DEGC %CU - 0.29 %NI - 0.55 FO - 1.09E19 LONGIT
A/W
0
0.1
0.2
0.3
0.4
0.5
0.6
0.7
0.8
0.9
1
0
10
20
30
40
50
60
70
80
90
100
110
220
220
TIME(MINUTES)

TRANSIENT NO. 25 [Z2TBV(ZP2TBV)]


IPTS OCONEE CLAD Z2TBV 10/20/83
HTC(W/M**2*K)
*10^2
0.0
2.0
4.0
6.0
8.0
10.0
12.0
14.0
16.0
18.0
20.0
TEMP.(DEG.C.)
30.0
60.0
90.0
120.0
150.0
180.0
210.0
240.0
270.0
300.0
330.0
PRESS.(MPA)
0.0
2.0
4.0
6.0
8.0
10.0
12.0
14.0
16.0
18.0
20.0
TIME(MIN.)
0.0
10.0
20.0
30.0
40.0
50.0
60.0
70.0
80.0
90.0
100.0
110.0
120.0
□ PRESS.(MPA)
○ TEMP.(DEG.C.)
△ HTC(W/M**2*K)

621

```
IPTS OCONEE CLAD Z2TBV  10/20/83                        1. FLAWS/M**3     FO =  1.090D+19

        -------------UNADJUSTED--------------           ---ADJUSTED----
WELD    P(F/E)      95%CI    %ERR   P(INITIA)   N*V      P(F/E)    %ERR    NTRIALS


  1     6.84D-03   6.18D-04   9.03   6.95D-03  0.040   2.74D-04              40000
  2     4.19D-03   3.96D-04   9.44   4.31D-03  0.040   1.68D-04              60000
  3     7.55D-03   6.48D-04   8.59   7.59D-03  0.010   7.55D-05              40000

                                            VESSEL     5.17D-04   5.81



 DEPTHS FOR INITIAL INITIATION (MM)
              3.2   6.3  12.7  19.0  25.4  38.1  50.8
  NUMBER      187   751   319   129    34    10     0
  PERCENT    13.1  52.5  22.3   9.0   2.4   0.7   0.0




TIMES OF FAILURE(MINUTES)
                  0.0  10.0  20.0  30.0  40.0  50.0  60.0  70.0  80.0  90.0 100.0 110.0 120.0
  NUMBER            0    78    23   171   228   187   205   146   124    98    63    85
  PERCENT         0.0   5.5   1.6  12.1  16.2  13.3  14.6  10.4   8.8   7.0   4.5   6.0

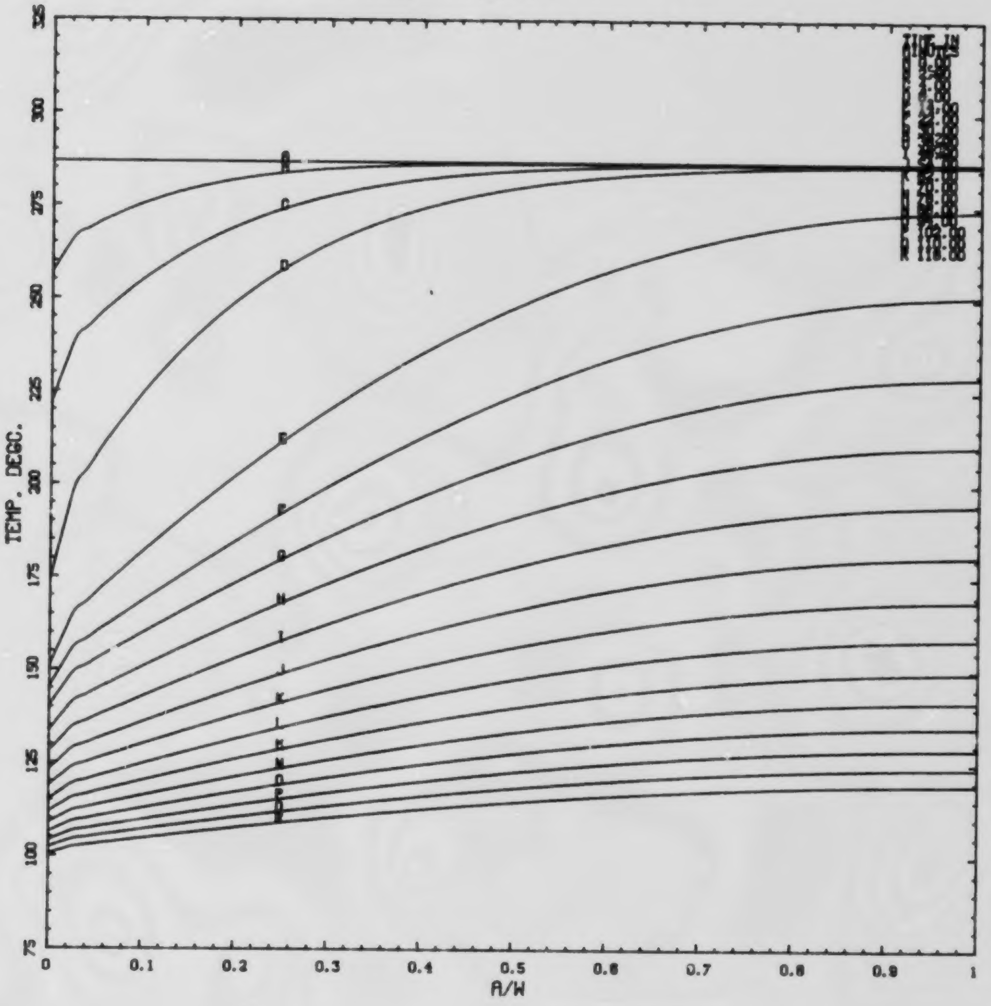



 INITIATION T-RTNDT(DEG.C)
              -55.6 -41.7 -27.8 -13.9   0.0  13.9  27.8  41.7  55.6  69.4  83.3  97.2 111.1
  NUMBER            0     3   117   495   583   219    32     3     0     0     0     0
  PERCENT         0.0   0.2   8.1  34.1  40.2  15.1   2.2   0.2   0.0   0.0   0.0   0.0



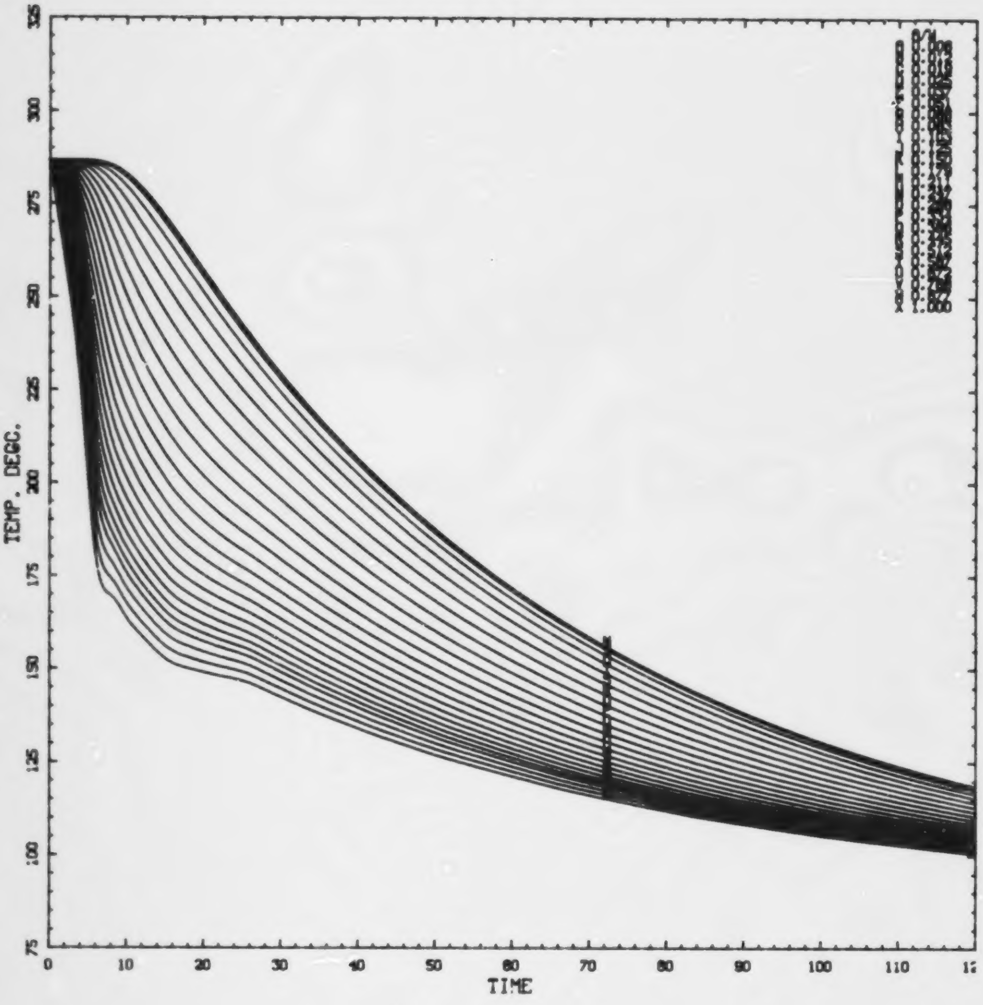

 ARREST T-RTNDT(DEG.C)
              -27.8 -13.9   0.0  13.9  27.8  41.7  55.6  69.4  83.3  97.2 111.1 125.0 138.9
  NUMBER            0     8    14     3     0     0     1     6    12     0     0     0
  PERCENT         0.0  18.2  31.8   6.8   0.0   0.0   2.3  13.6  27.3   0.0   0.0   0.0
```

IPTS OCONEE CLAD Z2TBV 10/20/83



623

IPTS OCONEE CLAD Z2TBV 10/20/83



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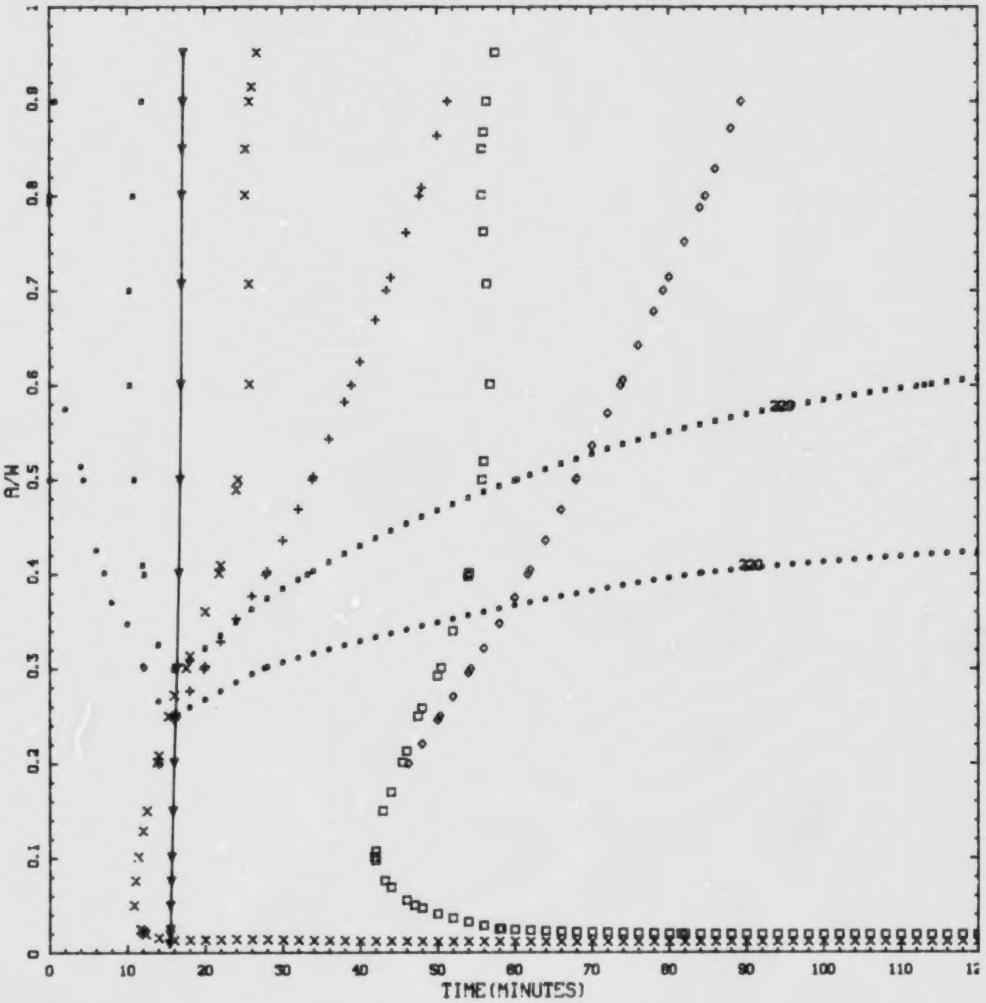
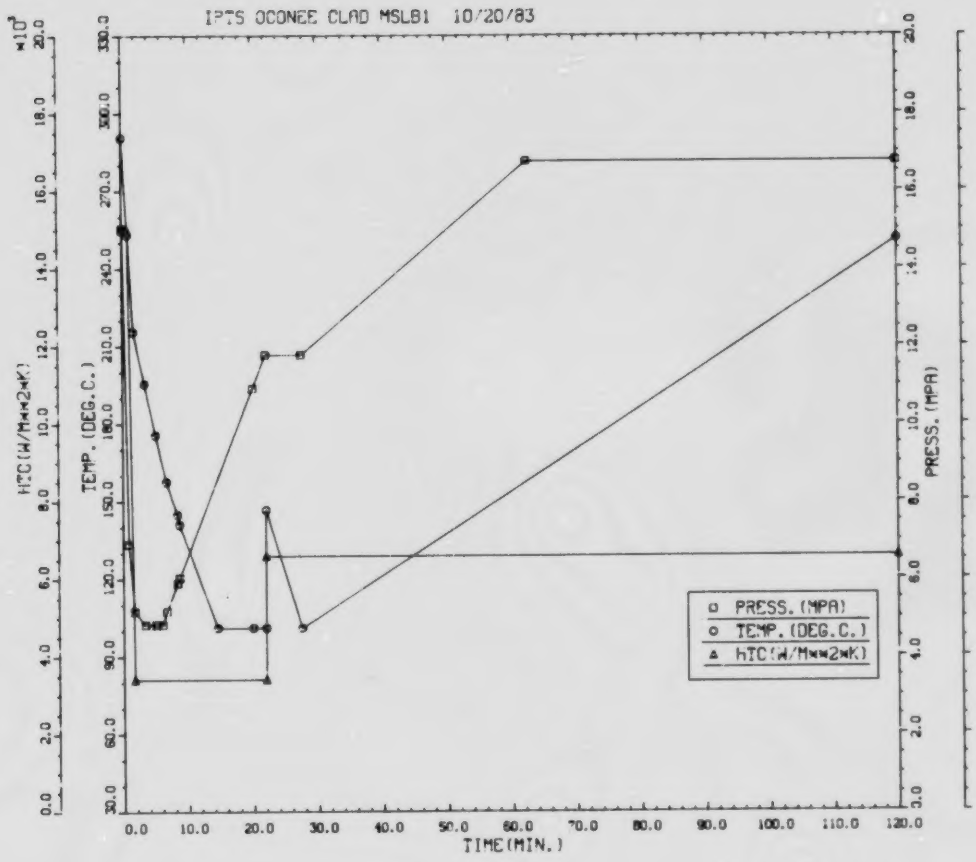
CRITICAL CRACK DEPTH CURVES FOR IPTS OCONEE CLAD Z2TBV 10/20/83
RTNDTO --6.7 DEGC %CU - 0.29 %NI - 0.55 FO - 1.09E19 LONGIT



625

TRANSIENT NO. 26 (MSLB1)

BEST COPY AVAILABLE


IPTS OCONEE CLAD MSLB1 10/20/83
HTC(W/M**2*K)
*10^3
0.0
2.0
4.0
6.0
8.0
10.0
12.0
14.0
16.0
18.0
20.0
TEMP.(DEG.C.)
30.0
60.0
90.0
120.0
150.0
180.0
210.0
240.0
270.0
300.0
330.0
PRESS.(MPA)
0.0
2.0
4.0
6.0
8.0
10.0
12.0
14.0
16.0
18.0
20.0
TIME(MIN.)
0.0
10.0
20.0
30.0
40.0
50.0
60.0
70.0
80.0
90.0
100.0
110.0
120.0
□ PRESS.(MPA)
○ TEMP.(DEG.C.)
▲ HTC(W/M**2*K)

```
IPTS OCONEE CLAD MSLB1  10/20/83                    1. FLAWS/M**3     FO =  1.090D+19

       -------------UNADJUSTED--------------            ---ADJUSTED----
WELD   P(F/E)       95%CI    %ERR   P(INITIA)   N*V    P(F/E)    %ERR    NTRIALS


 1     8.56D-03   7.96D-04   9.31   3.16D-02  0.040   3.42D-04              30000
 2     4.70D-03   4.59D-04   9.76   2.03D-02  0.040   1.88D-04              50000
 3     8.95D-03   8.14D-04   9.10   3.36D-02  0.010   8.95D-05              30000

                                           VESSEL     6.20D-04   6.08


 DEPTHS FOR INITIAL INITIATION (MM)
              3.2   6.3  12.7  19.0  25.4  38.1  50.8
   NUMBER    2507  1817   605   120    14     1     0
   PERCENT   49.5  35.9  11.9   2.4   0.3   0.0   0.0




TIMES OF FAILURE(MINUTES)
               0.0  10.0  20.0  30.0  40.0  50.0  60.0  70.0  80.0  90.0 100.0 110.0 120.0
   NUMBER          0     0   490   753    41     6     4     0     0     0     0     0
   PERCENT       0.0   0.0  37.9  58.2   3.2   0.5   0.3   0.0   0.0   0.0   0.0   0.0

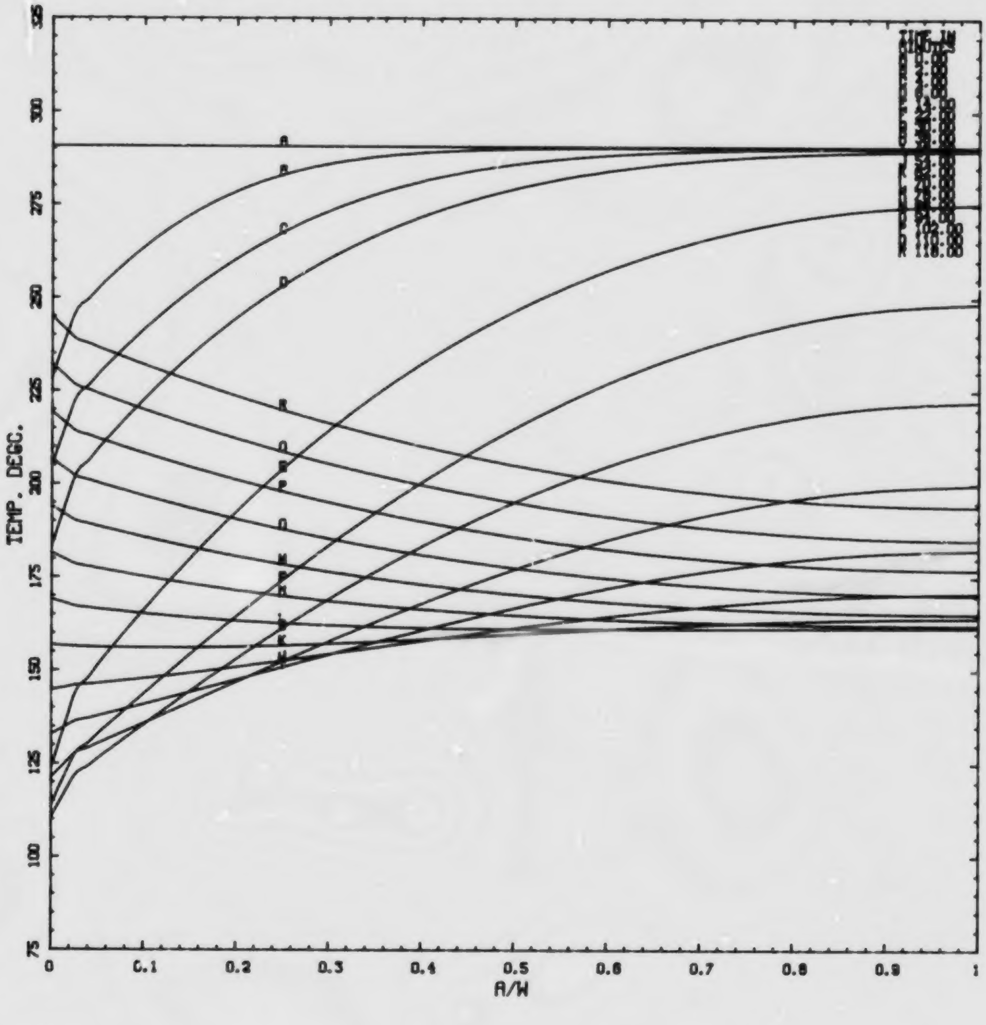



 INITIATION T-RTNDT(DEG.C)
            -55.6 -41.7 -27.8 -13.9   0.0  13.9  27.8  41.7  55.6  69.4  83.3  97.2 111.1
   NUMBER          1    13   283  1389  2200  1085   697   374    22     0     0     0
   PERCENT       0.0   0.2   4.7  22.9  36.3  17.9  11.5   6.2   0.4   0.0   0.0   0.0

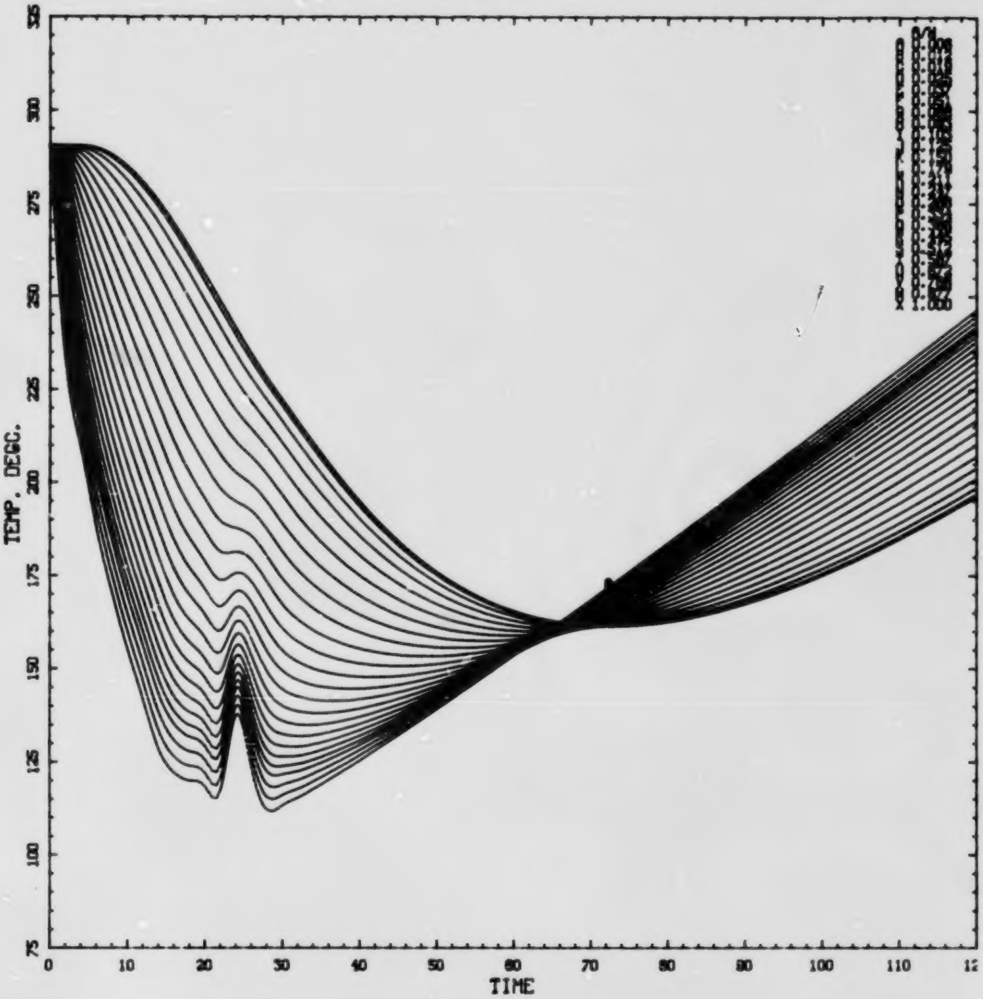



 ARREST T-RTNDT(DEG.C)
            -27.8 -13.9   0.0  13.9  27.8  41.7  55.6  69.4  83.3  97.2 111.1 125.0 138.9
   NUMBER          0     0     0     1     2     1    47  1269  3245   176     0    29
   PERCENT       0.0   0.0   0.0   0.0   0.0   0.0   1.0  26.6  68.0   3.7   0.0   0.6
```

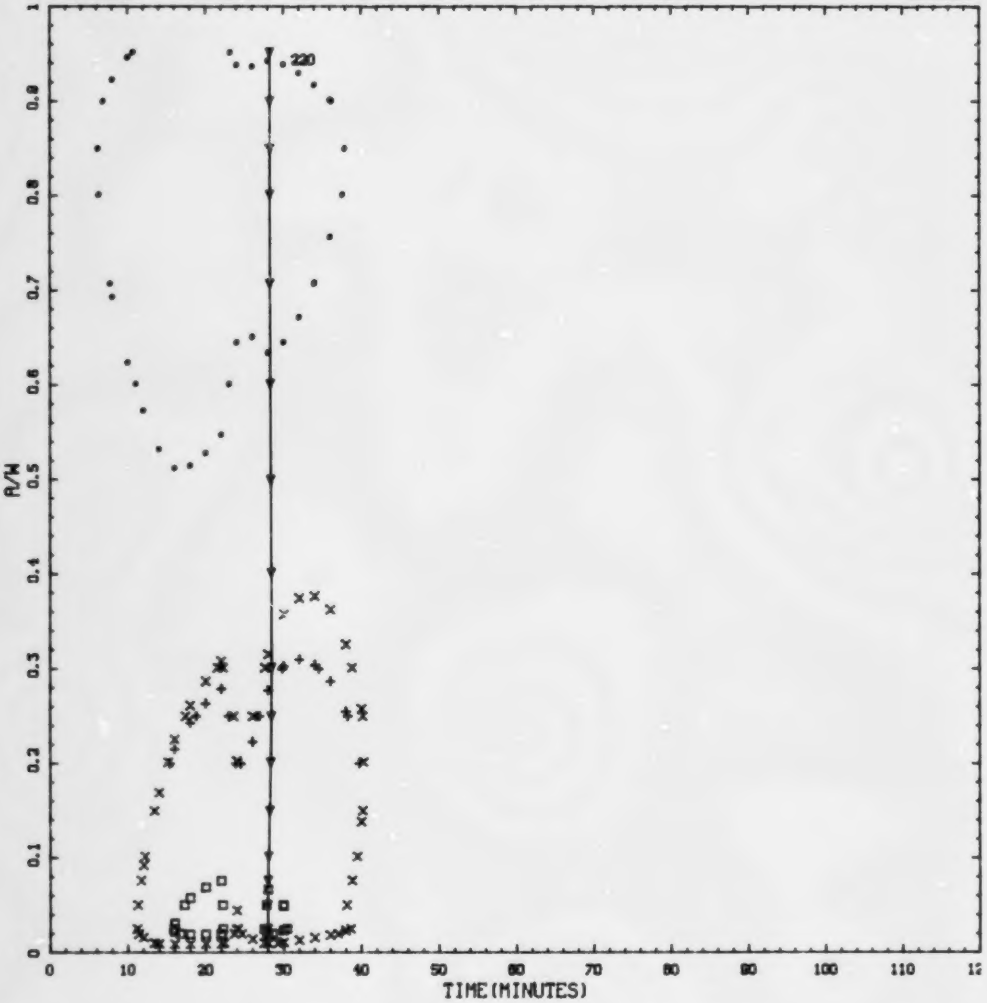
IPTS OCONEE CLAD MSLB1 10/20/83



IPTS OCONEE CLAD MSLB1 10/20/83

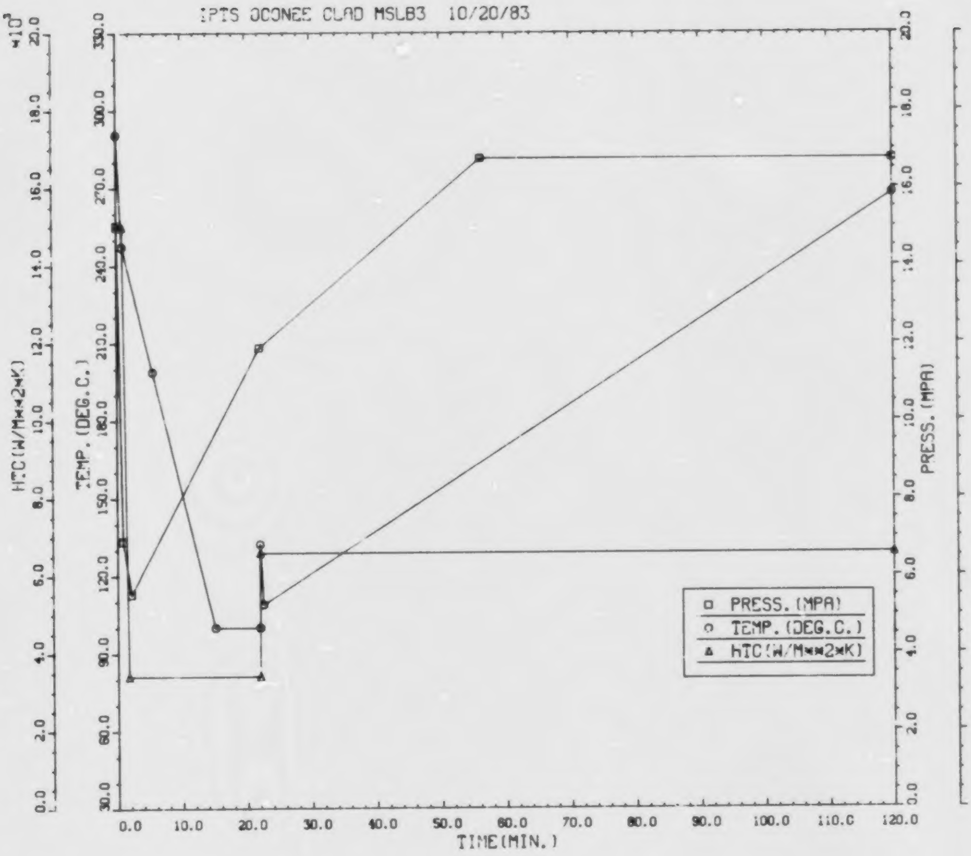


CRITICAL CRACK DEPTH CURVES FOR IPTS OCONEE CLAD MS.B1 10/20/83
RTNDTO --6.7 DEGC %CU - 0.29 %NI - 0.55 FO - 1.09E19 LONGIT



TRANSIENT NO. 28 (MSLB3)

632


IPTS OCONEE CLAD MSLB3 10/20/83
HTC(W/M**2*K) *10³
0.0
2.0
4.0
6.0
8.0
10.0
12.0
14.0
16.0
18.0
20.0
TEMP.(DEG.C.)
30.0
60.0
90.0
120.0
150.0
180.0
210.0
240.0
270.0
300.0
330.0
PRESS.(MPA)
0.0
2.0
4.0
6.0
8.0
10.0
12.0
14.0
16.0
18.0
20.0
TIME(MIN.)
0.0
10.0
20.0
30.0
40.0
50.0
60.0
70.0
80.0
90.0
100.0
110.0
120.0
□ PRESS.(MPA)
○ TEMP.(DEG.C.)
▵ HTC(W/M**2*K)

```
IPTS OCONEE CLAD MSLB3  10/20/83                          1. FLAWS/M**3     FO =  1.090D+19

        -------------UNADJUSTED--------------           ---ADJUSTED----
WELD    P(F/E)      95%CI     %ERR   P(INITIA)   N*V     P(F/E)     %ERR     NTRIALS


  1     5.30D-03   4.87D-04   9.19   3.28D-02   0.040   2.12D-04              50000
  2     3.17D-03   2.98D-04   9.40   1.94D-02   0.040   1.27D-04              80000
  3     6.36D-03   5.96D-04   9.37   3.48D-02   0.010   6.36D-05              40000

                                          VESSEL        4.02D-04   5.87


DEPTHS FOR INITIAL INITIATION (MM)
             3.2   6.3  12.7  19.0  25.4  38.1  50.8
  NUMBER    4126  2668   847   143    14     1     0
  PERCENT   52.9  34.2  10.9   1.8   0.2   0.0   0.0



TIMES OF FAILURE(MINUTES)
                0.0  10.0  20.0  30.0  40.0  50.0  60.0  70.0  80.0  90.0 100.0 110.0 120.0
  NUMBER              0   100  1051   156     7     1     1     0     0     0     0     0
  PERCENT           0.0   7.6  79.9  11.9   0.5   0.1   0.1   0.0   0.0   0.0   0.0   0.0

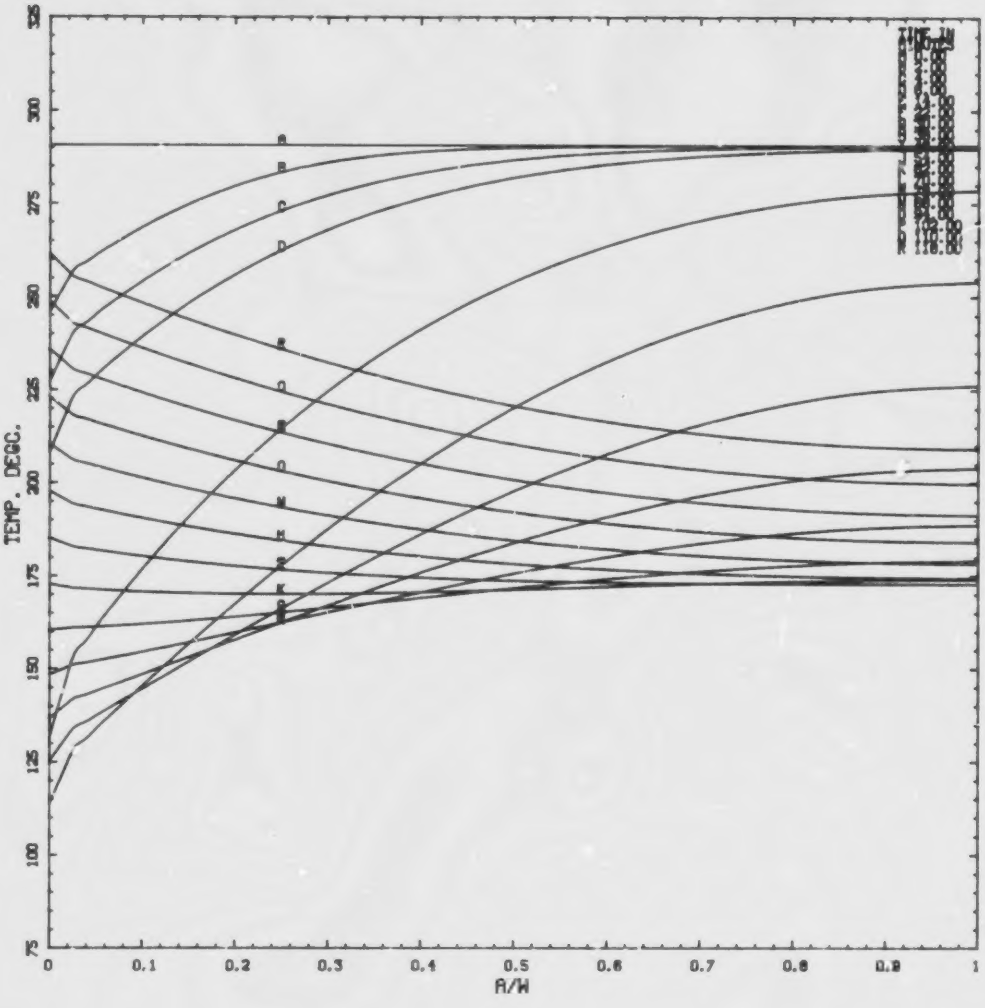


INITIATION T-RTNDT(DEG.C)
             -55.6 -41.7 -27.8 -13.9   0.0  13.9  27.8  41.7  55.6  69.4  83.3  97.2 111.1
  NUMBER               1    39   421  2111  3348  1679   400   304    13     0     0     0
  PERCENT            0.0   0.5   5.1  25.4  40.3  20.2   4.8   3.7   0.2   0.0   0.0   0.0

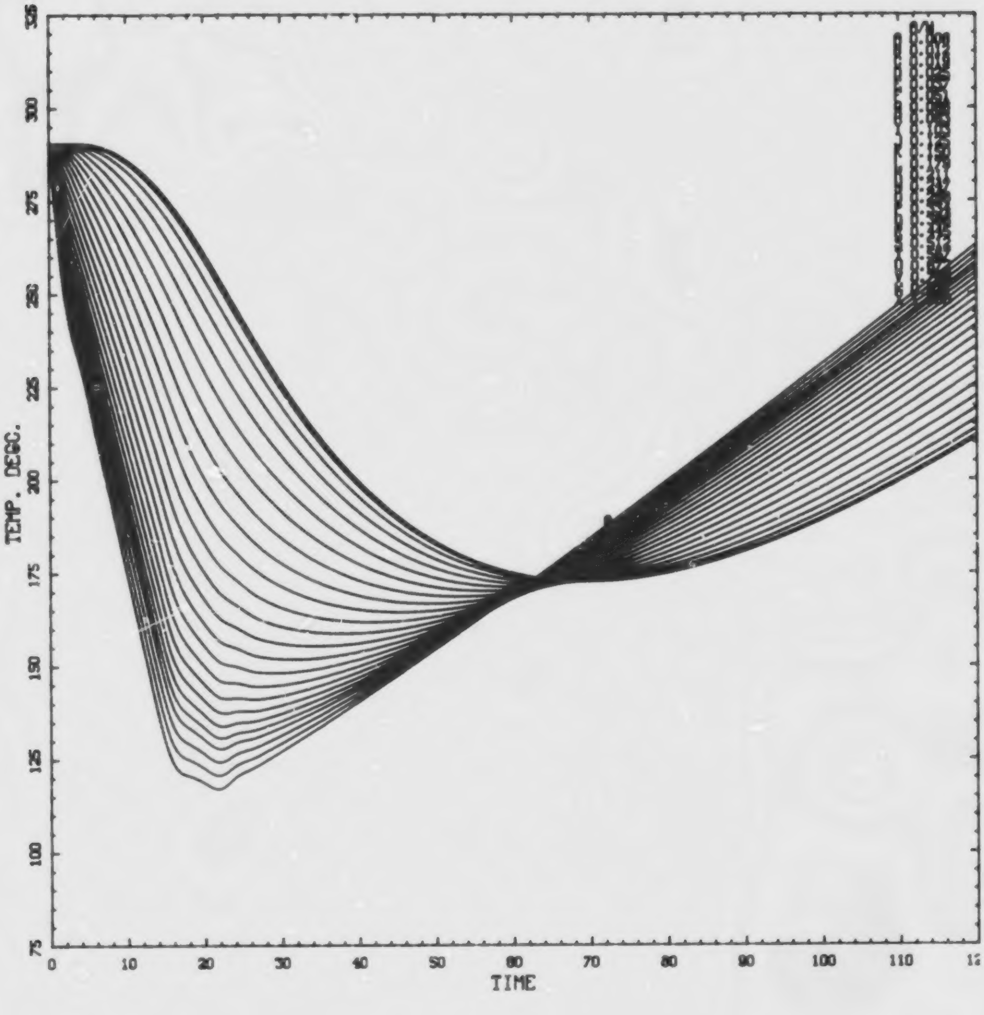


ARREST T-RTNDT(DEG.C)
             -27.8 -13.9   0.0  13.9  27.8  41.7  55.6  69.4  83.3  97.2 111.1 125.0 138.9
  NUMBER               0     0     0     0     1     2    44  1440  5138   373     0     2
  PERCENT            0.0   0.0   0.0   0.0   0.0   0.0   0.6  20.6  73.4   5.3   0.0   0.0
```

IPTS OCONEE CLAD MSLB3 10/20/83

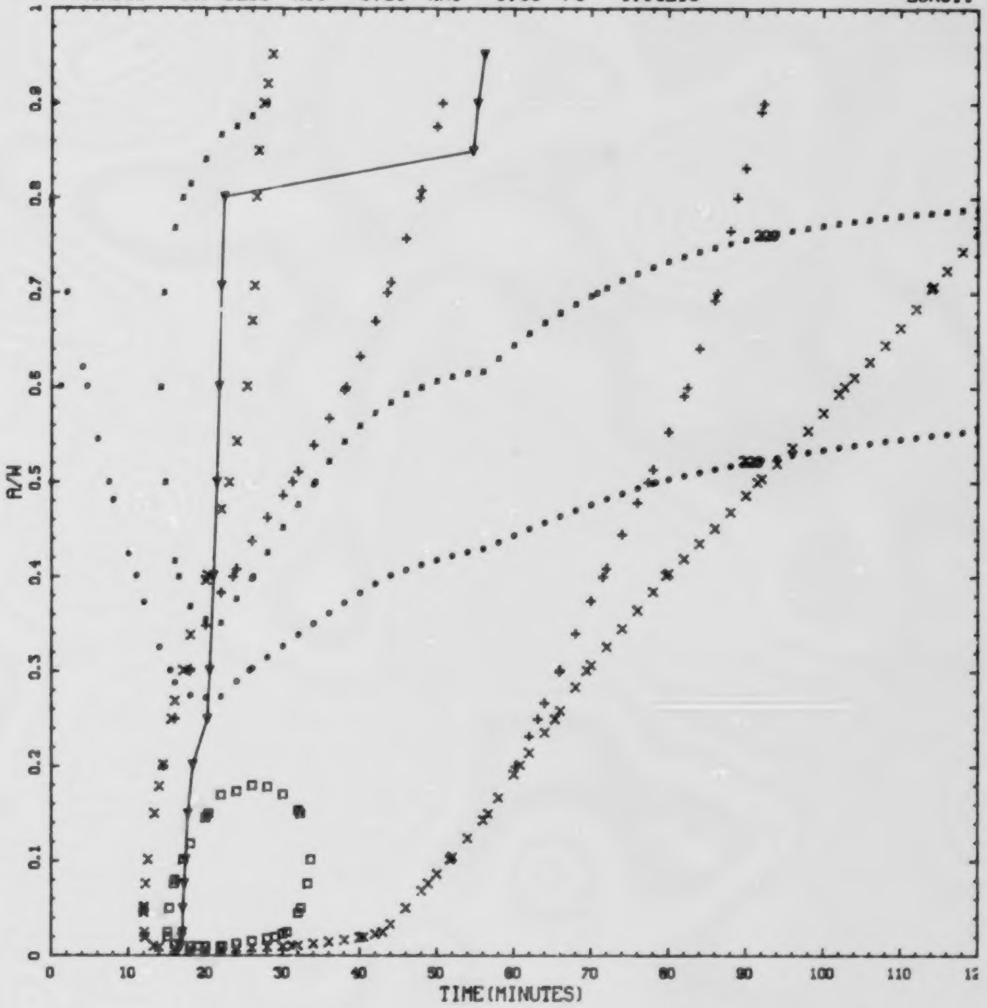


635

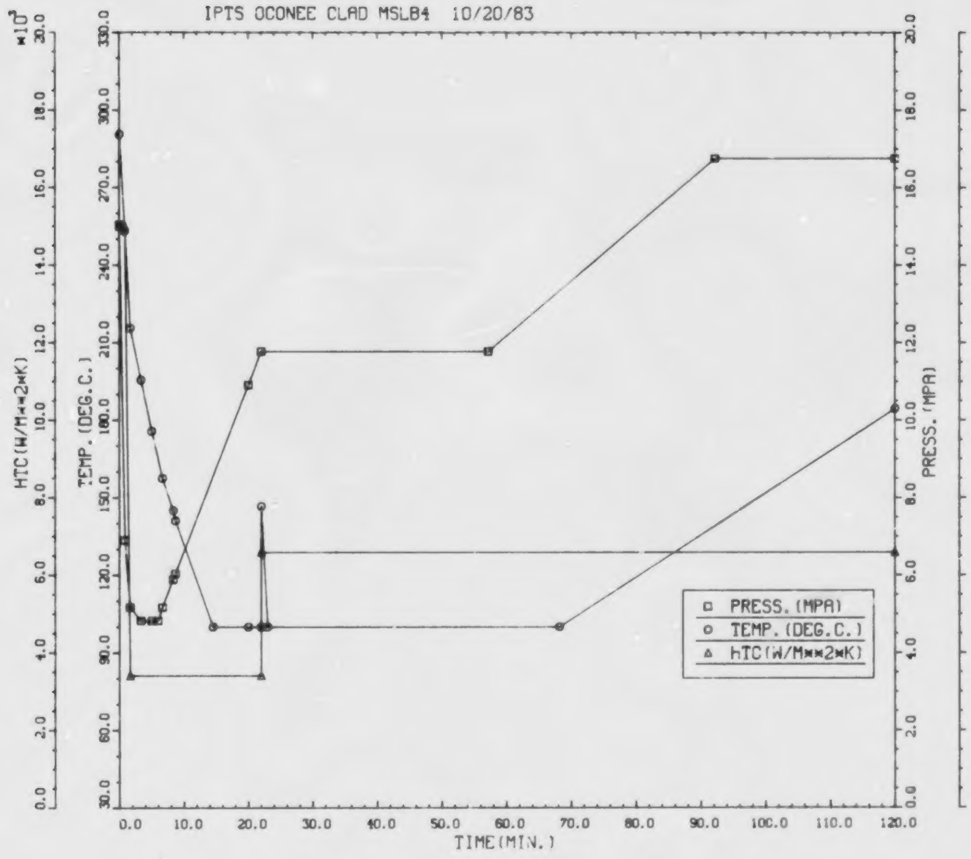


IPTS OCONEE CLAD MSLB3 10/20/83

CRITICAL CRACK DEPTH CURVES FOR IPTS OCONEE CLAD MSLB3 10/20/83
RTNDTO --6.7 DEGC %CU - 0.29 %NI - 0.55 FO - 1.09E19 LONGIT



TRANSIENT NO. 29 (MSLB4)


IPTS OCONEE CLAD MSLB4 10/20/83
HTC(W/M**2*K)
*10³
0.0
2.0
4.0
6.0
8.0
10.0
12.0
14.0
16.0
18.0
20.0
TEMP.(DEG.C.)
30.0
60.0
90.0
120.0
150.0
180.0
210.0
240.0
270.0
300.0
330.0
PRESS.(MPA)
0.0
2.0
4.0
6.0
8.0
10.0
12.0
14.0
16.0
18.0
20.0
TIME(MIN.)
0.0
10.0
20.0
30.0
40.0
50.0
60.0
70.0
80.0
90.0
100.0
110.0
120.0
□ PRESS.(MPA)
○ TEMP.(DEG.C.)
△ HTC(W/M**2*K)

```
IPTS OCONEE CLAD MSLB4  10/20/83                       1. FLAWS/M**3     F0 =  1.090D+19

        -------------UNADJUSTED--------------          ---ADJUSTED----
WELD    P(F/E)      95%CI    %ERR   P(INITIA)   N*V     P(F/E)    %ERR    NTRIALS


 1      3.98D-02   2.89D-03   7.27   4.79D-02  0.040  1.59D-03                10000
 2      2.35D-02   2.26D-03   9.60   3.11D-02  0.040  9.40D-04                10000
 3      4.14D-02   2.95D-03   7.12   5.13D-02  0.010  4.14D-04                10000

                                            VESSEL     2.95D-03   5.08


 DEPTHS FOR INITIAL INITIATION (MM)
                3.2   6.3  12.7  19.0  25.4  38.1  50.8
   NUMBER    1137   757   259    50    13     2     0
   PERCENT   51.3  34.1  11.7   2.3   0.6   0.1   0.0



TIMES OF FAILURE(MINUTES)
                0.0  10.0  20.0  30.0  40.0  50.0  60.0  70.0  80.0  90.0 100.0 110.0 120.0
   NUMBER           0     0   536   421   183   211   320    94    17     1     0     0
   PERCENT        0.0   0.0  30.1  23.6  10.3  11.8  17.9   5.3   1.0   0.1   0.0   0.0

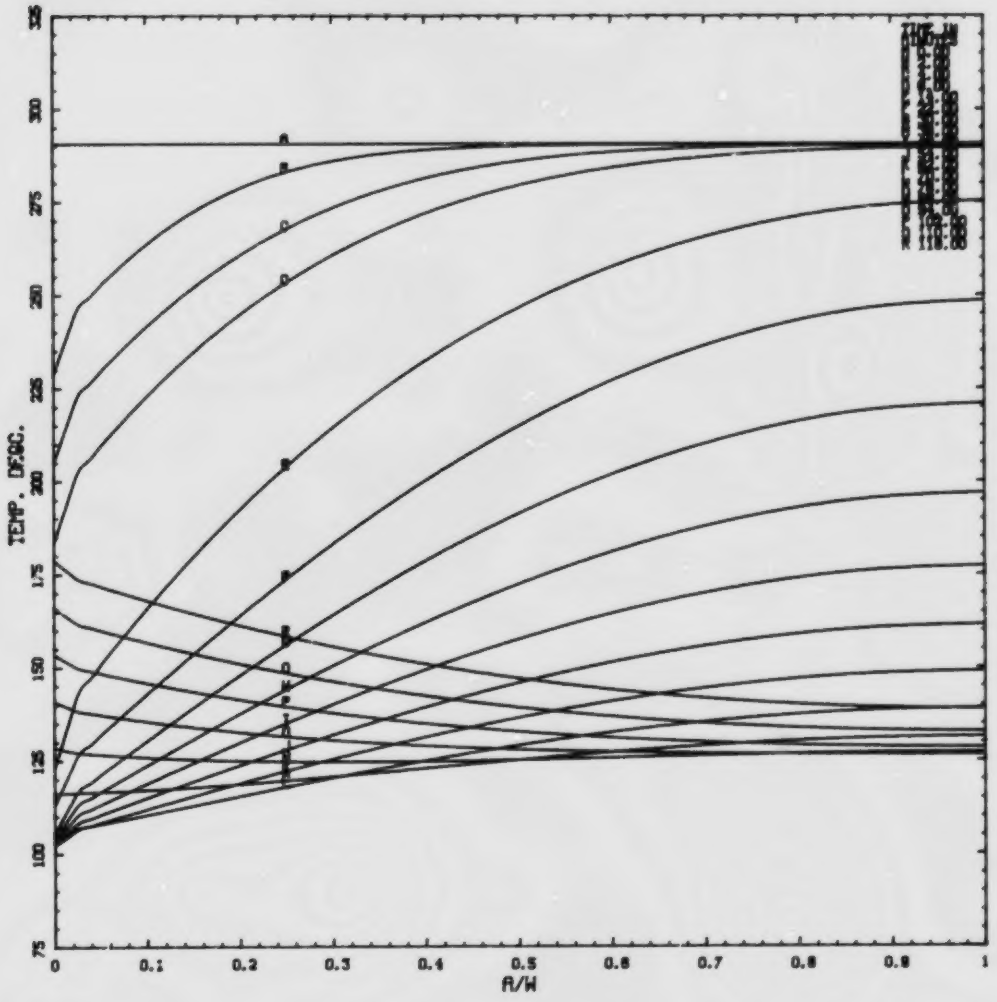


 INITIATION T-RTNDT(DEG.C)
              -55.6 -41.7 -27.8 -13.9   0.0  13.9  27.8  41.7  55.6  69.4  83.3  97.2 111.1
   NUMBER           0    22   171   610   977   810   854   292     8     0     0     0
   PERCENT        0.0   0.6   4.6  16.3  26.1  21.6  22.8   7.8   0.2   0.0   0.0   0.0

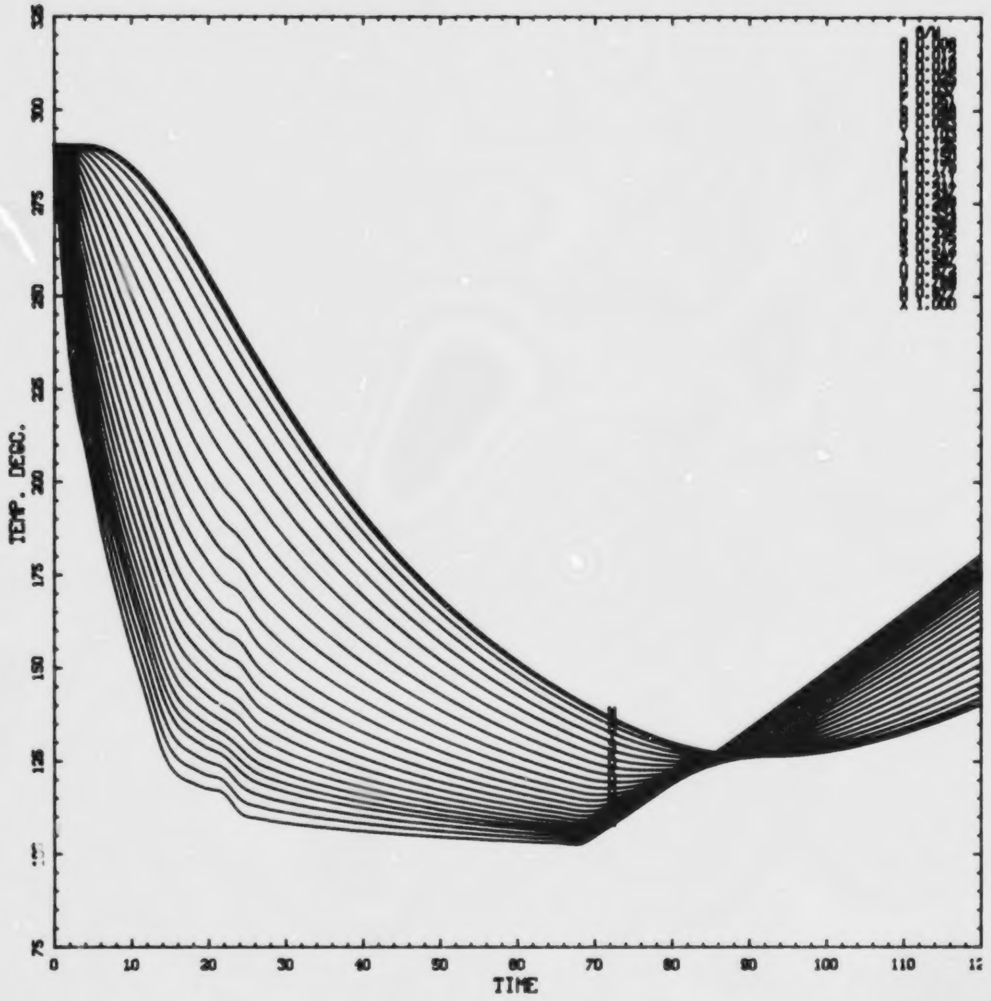


 ARREST T-RTNDT(DEG.C)
              -27.8 -13.9   0.0  13.9  27.8  41.7  55.6  69.4  83.3  97.2 111.1 125.0 138.9
   NUMBER           0     0     0     0     2     0    11   427  1451    55     0    15
   PERCENT        0.0   0.0   0.0   0.0   0.1   0.0   0.6  21.8  74.0   2.8   0.0   0.8
```

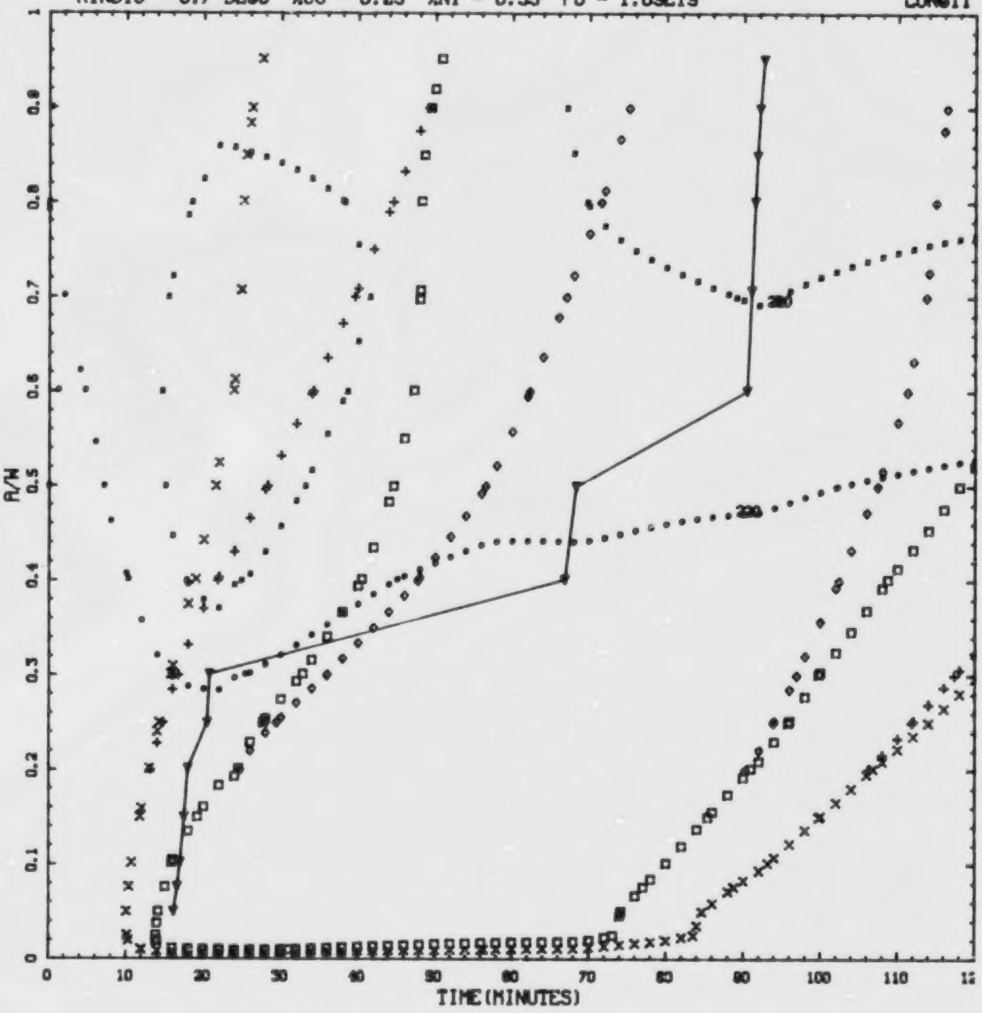
IPTS OCONEE CLAD MSLB4 10/20/83



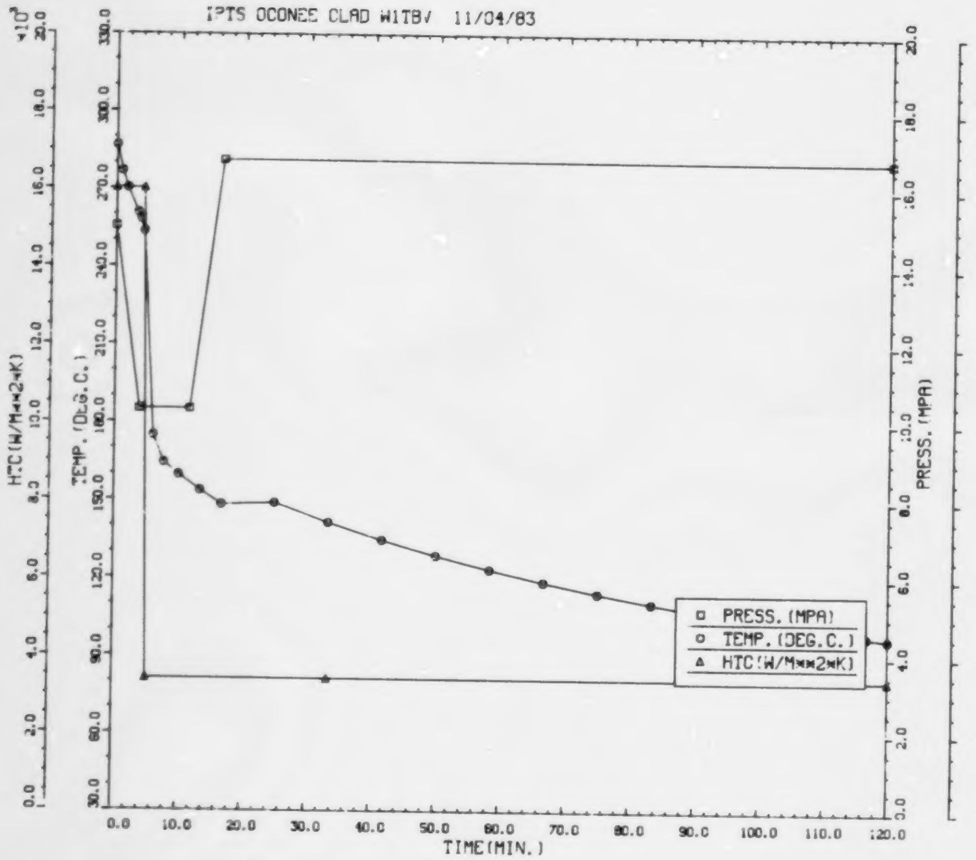
IPTS OCONEE CLAD MSLB4 10/20/83



CRITICAL CRACK DEPTH CURVES FOR IPTS OCONEE CLAD MSLB4 10/20/83
RTNDTO --6.7 DEGC %CU - 0.29 %NI - 0.55 FO - 1.09E19 LONGIT



TRANSIENT NO. 33 [W1TBV(ZW1TBV)]


IPTS OCONEE CLAD W1TBV 11/04/83
HTC(W/M**2*K) $\times 10^3$
0.0
2.0
4.0
6.0
8.0
10.0
12.0
14.0
16.0
18.0
20.0
TEMP.(DEG.C.)
30.0
60.0
90.0
120.0
150.0
180.0
210.0
240.0
270.0
300.0
330.0
PRESS.(MPA)
0.0
2.0
4.0
6.0
8.0
10.0
12.0
14.0
16.0
18.0
20.0
TIME(MIN.)
0.0
10.0
20.0
30.0
40.0
50.0
60.0
70.0
80.0
90.0
100.0
110.0
120.0
□ PRESS.(MPA)
○ TEMP.(DEG.C.)
△ HTC(W/M**2*K)

```
IPTS OCONEE CLAD W1TBV  11/04/83                          1. FLAWS/M**3     F0 =  1.090D+19

         -------------UNADJUSTED--------------            ---ADJUSTED----
WELD     P(F/E)      95%CI     %ERR   P(INITIA)    N*V     P(F/E)    %ERR     NTRIALS


 1       8.48D-03   4.14D-04   4.88   8.68D-03  0.040   3.39D-04             110000
 2       5.44D-03   2.67D-04   4.92   5.62D-03  0.040   2.18D-04             170000
 3       9.30D-03   4.54D-04   4.89   9.48D-03  0.010   9.30D-05             100000

                                              VESSEL    6.50D-04   3.11



 DEPTHS FOR INITIAL INITIATION (MM)
                  3.2   6.3  12.7  19.0  25.4  38.1  50.8
   NUMBER        689  2571   994   440   150    21     2
   PERCENT      14.2  52.8  20.4   9.0   3.1   0.4   0.0




TIMES OF FAILURE(MINUTES)
                   0.0  10.0  20.0  30.0  40.0  50.0  60.0  70.0  80.0  90.0 100.0 110.0 120.0
   NUMBER                  0    16     9    72   195   343   484   603   661   704   780   880
   PERCENT               0.0   0.3   0.2   1.5   4.1   7.2  10.2  12.7  13.9  14.8  16.4  18.5

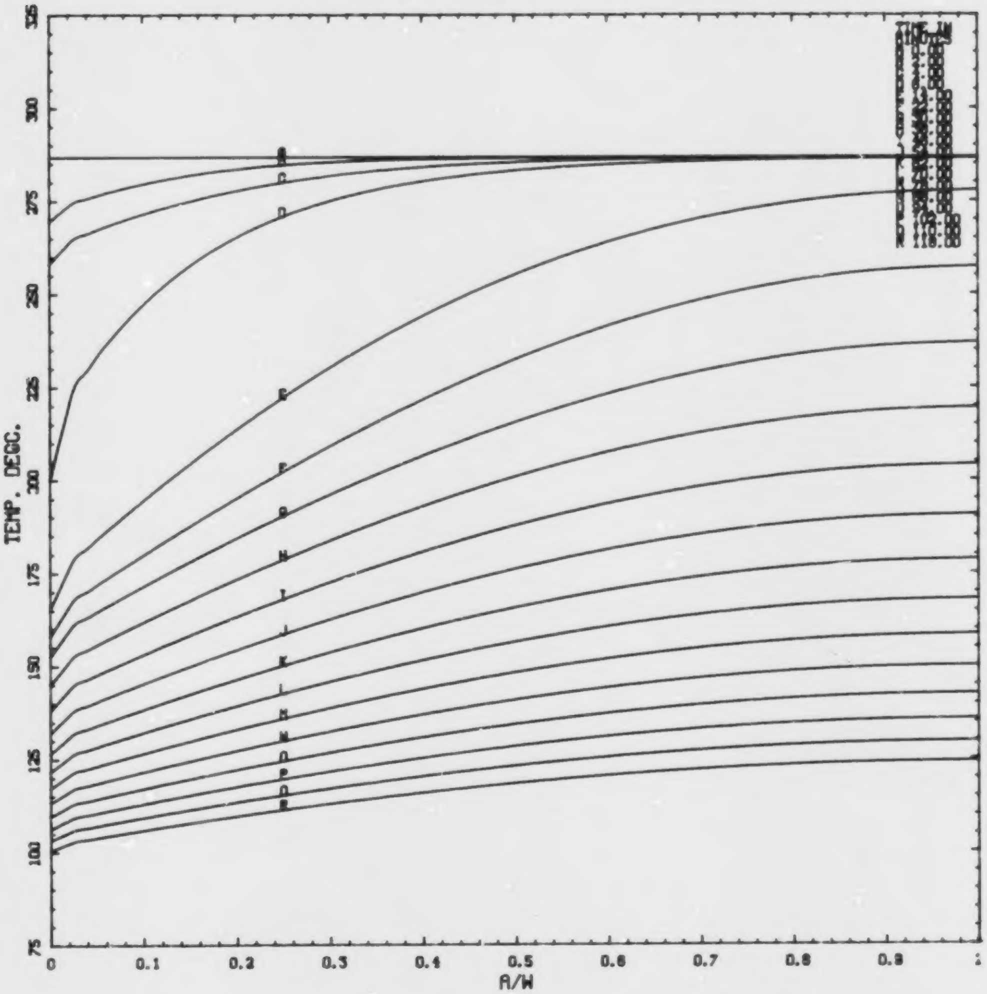



 INITIATION T-RTNDT(DEG.C)
                 -55.6 -41.7 -27.8 -13.9   0.0  13.9  27.8  41.7  55.6  69.4  83.3  97.2 111.1
   NUMBER                  2   155   988  1948  1433   328    24     5     0     0     0     0
   PERCENT               0.0   3.2  20.2  39.9  29.3   6.7   0.5   0.1   0.0   0.0   0.0   0.0



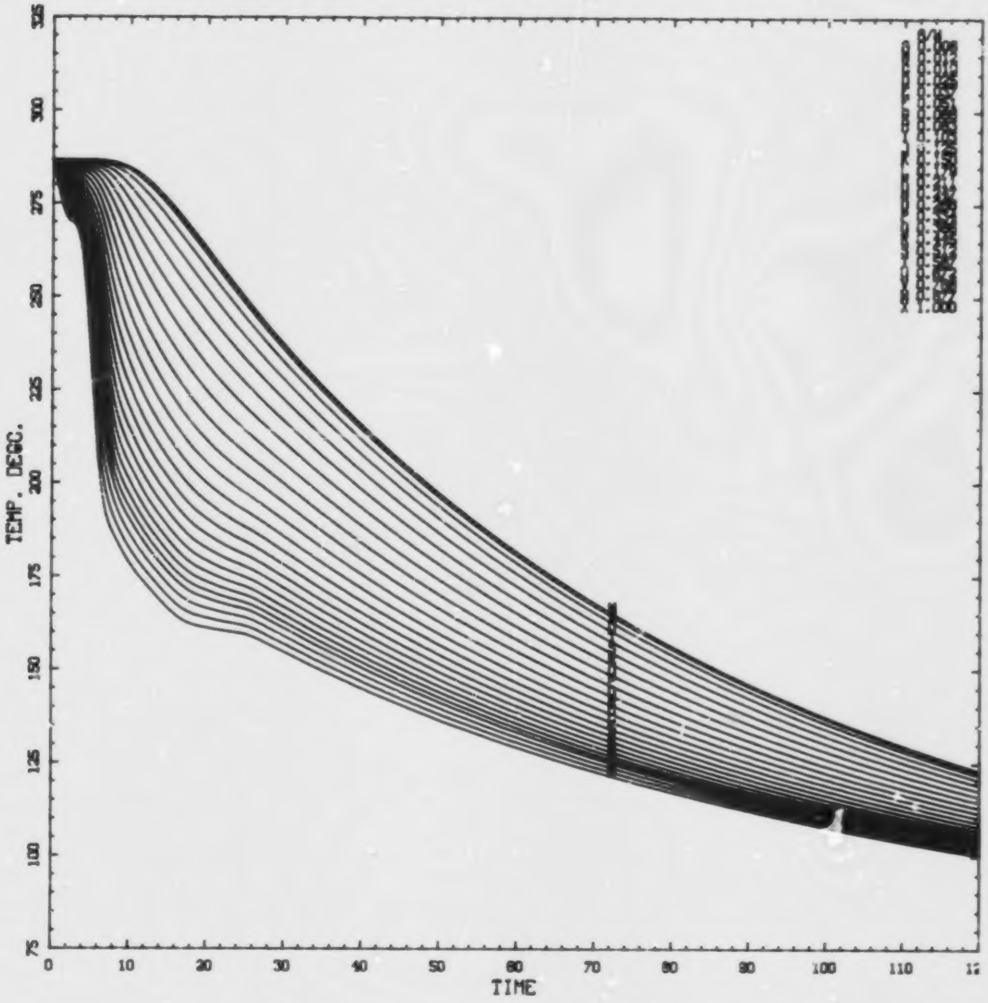

 ARREST T-RTNDT(DEG.C)
                 -27.8 -13.9   0.0  13.9  27.8  41.7  55.6  69.4  83.3  97.2 111.1 125.0 138.9
   NUMBER                  1    18    81    25     0     0     0     0    11     0     0     0
   PERCENT               0.7  13.2  59.6  18.4   0.0   0.0   0.0   0.0   8.1   0.0   0.0   0.0
```

IPTS OCONEE CLAD W1TBV 11/04/83



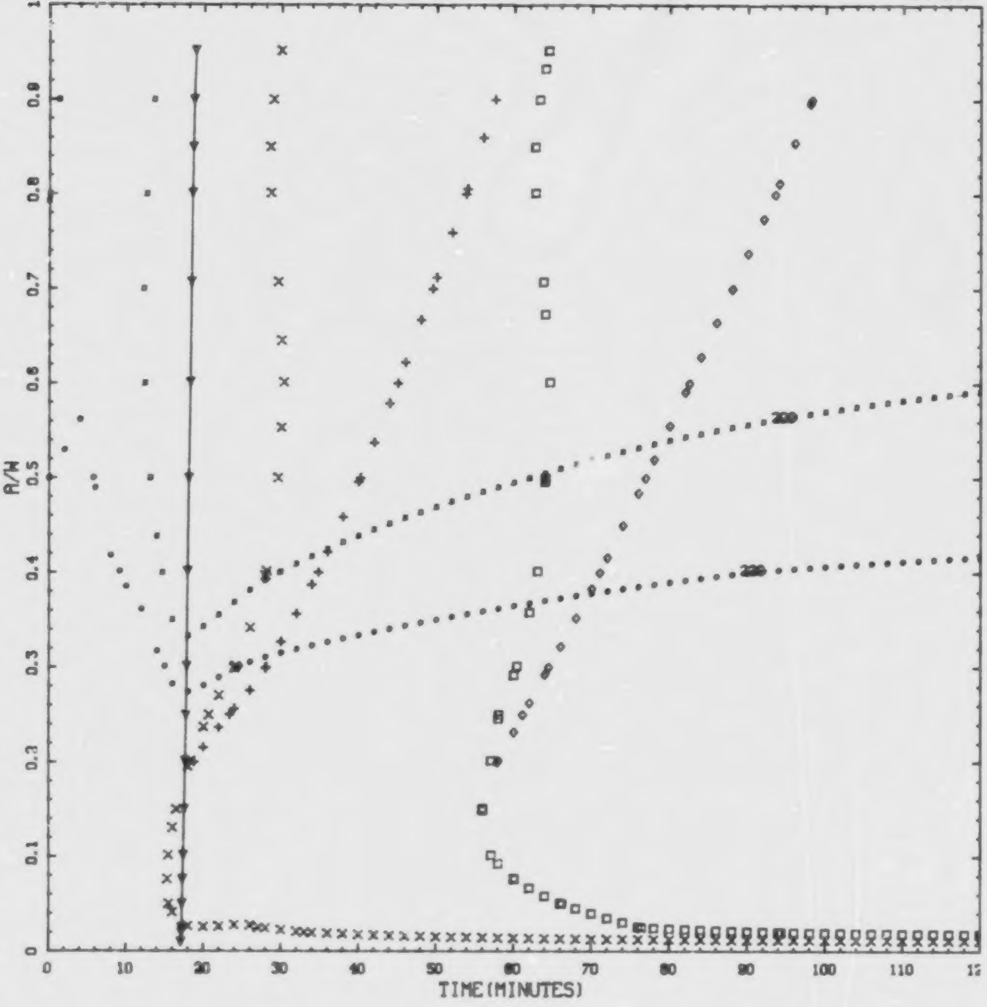
647

IPTS OCONEE CLAD W1TBV 11/04/83



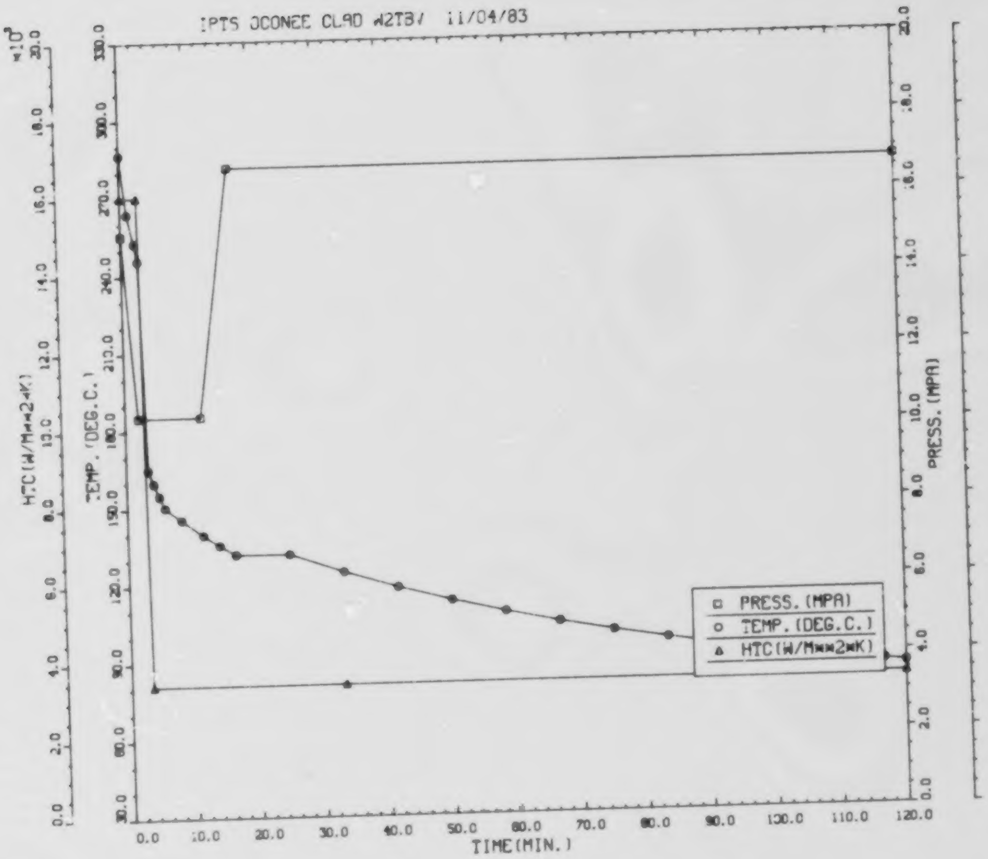
648

CRITICAL CRACK DEPTH CURVES FOR IPTS OCONEE CLAD W1TBV 11/04/83
RTNDTO =-6.7 DEGC %CU = 0.29 %NI = 0.55 FO = 1.09E19 LONGIT



649

TRANSIENT NO. 34 [W2TBV(ZW2TBV)]


IPTS OCONEE CLAD W2TBV 11/04/83
HTC(W/M**2*K) ×10³
0.0
2.0
4.0
6.0
8.0
10.0
12.0
14.0
16.0
18.0
20.0
TEMP. (DEG.C.)
30.0
60.0
90.0
120.0
150.0
180.0
210.0
240.0
270.0
300.0
330.0
PRESS. (MPA)
0.0
2.0
4.0
6.0
8.0
10.0
12.0
14.0
16.0
18.0
20.0
TIME(MIN.)
0.0
10.0
20.0
30.0
40.0
50.0
60.0
70.0
80.0
90.0
100.0
110.0
120.0
□ PRESS. (MPA)
○ TEMP. (DEG.C.)
△ HTC(W/M**2*K)

651

```
IPTS OCONEE CLAD W2TBV  11/04/83                         1. FLAWS/M**3     F0 =  1.090D+19

       -------------UNADJUSTED--------------              ---ADJUSTED----
WELD    P(F/E)      95%CI     %ERR   P(INITIA)   N*V     P(F/E)    %ERR     NTRIALS


  1     2.35D-02   2.26D-03   9.60   2.37D-02  0.040   9.40D-04             10000
  2     1.53D-02   1.30D-03   8.47   1.54D-02  0.040   6.12D-04             20000
  3     2.40D-02   2.28D-03   9.50   2.46D-02  0.010   2.40D-04             10000

                                        VESSEL      1.79D-03   5.95



 DEPTHS FOR INITIAL INITIATION (MM)
                 3.2   6.3  12.7  19.0  25.4  38.1  50.8
   NUMBER        240   731   256    89    29     4     0
   PERCENT      17.8  54.2  19.0   6.6   2.1   0.3   0.0



TIMES OF FAILURE(MINUTES)
                 0.0  10.0  20.0  30.0  40.0  50.0  60.0  70.0  80.0  90.0 100.0 110.0 120.0
   NUMBER          0    31    25    90   138   146   171   170   173   146   122   117
   PERCENT       0.0   2.3   1.9   6.8  10.4  11.0  12.9  12.8  13.0  11.0   9.2   8.8

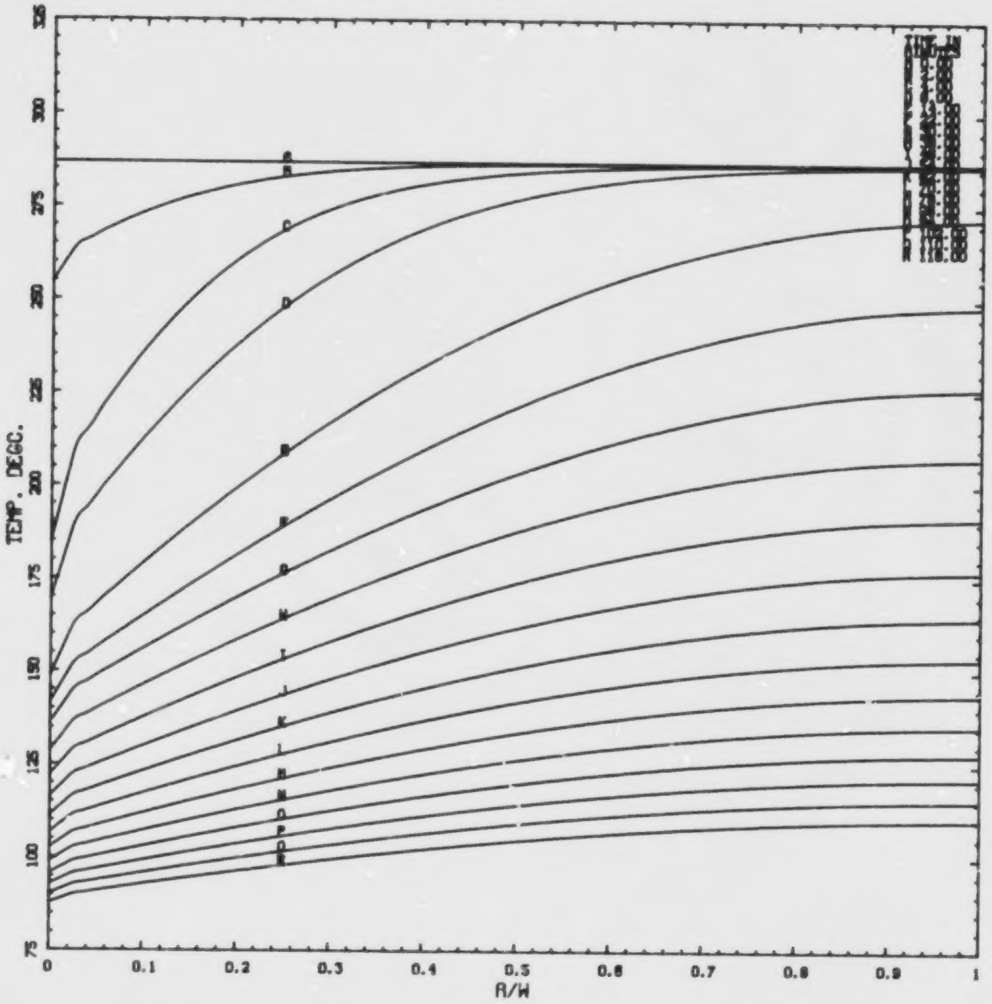


 INITIATION T-RTNDT(DEG.C)
             -55.6 -41.7 -27.8 -13.9   0.0  13.9  27.8  41.7  55.6  69.4  83.3  97.2 111.1
   NUMBER          0    37   226   541   429   108    18     2     1     0     0     0
   PERCENT       0.0   2.7  16.6  39.7  31.5   7.9   1.3   0.1   0.1   0.0   0.0   0.0

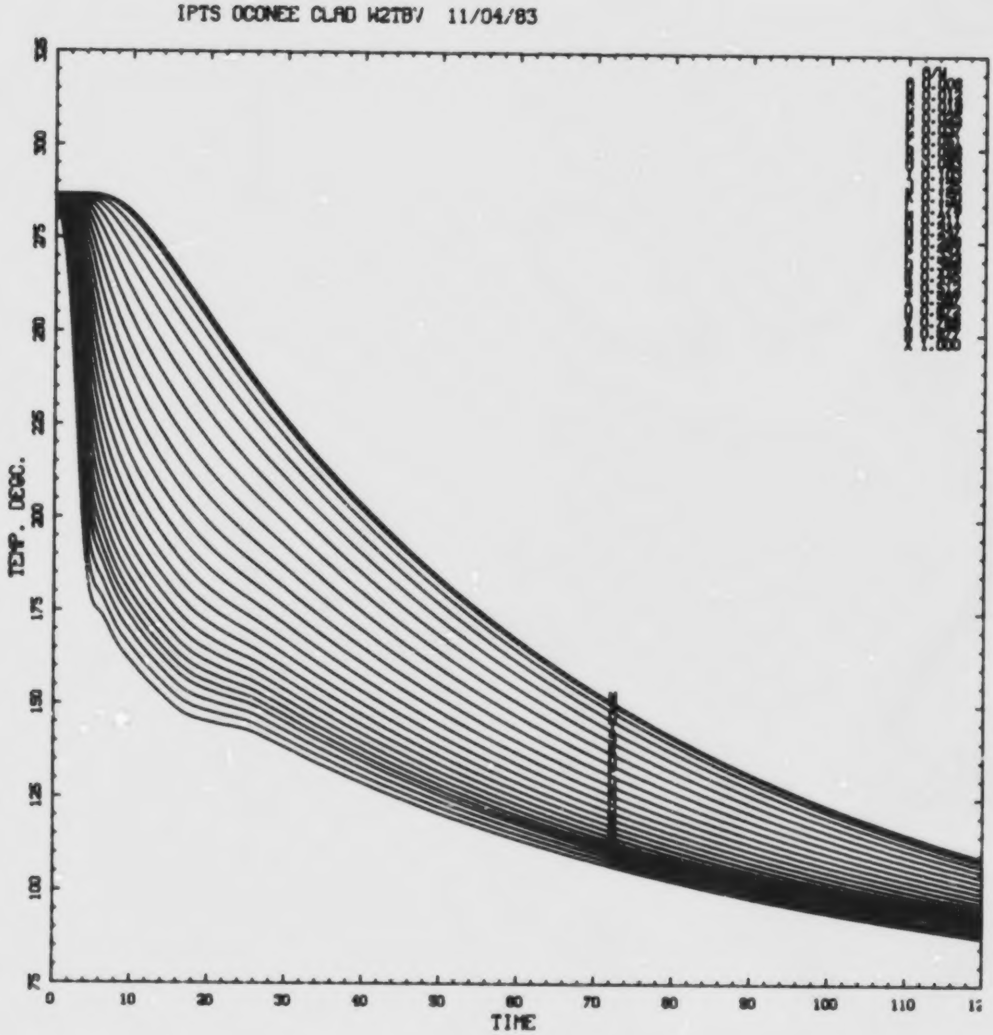


 ARREST T-RTNDT(DEG.C)
             -27.8 -13.9   0.0  13.9  27.8  41.7  55.6  69.4  83.3  97.2 111.1 125.0 138.9
   NUMBER          0     1    15     4     1     0     0     3     9     0     0     0
   PERCENT       0.0   3.0  45.5  12.1   3.0   0.0   0.0   9.1  27.3   0.0   0.0   0.0
```

IPTS OCONEE CLAD W2TBV 11/04/83

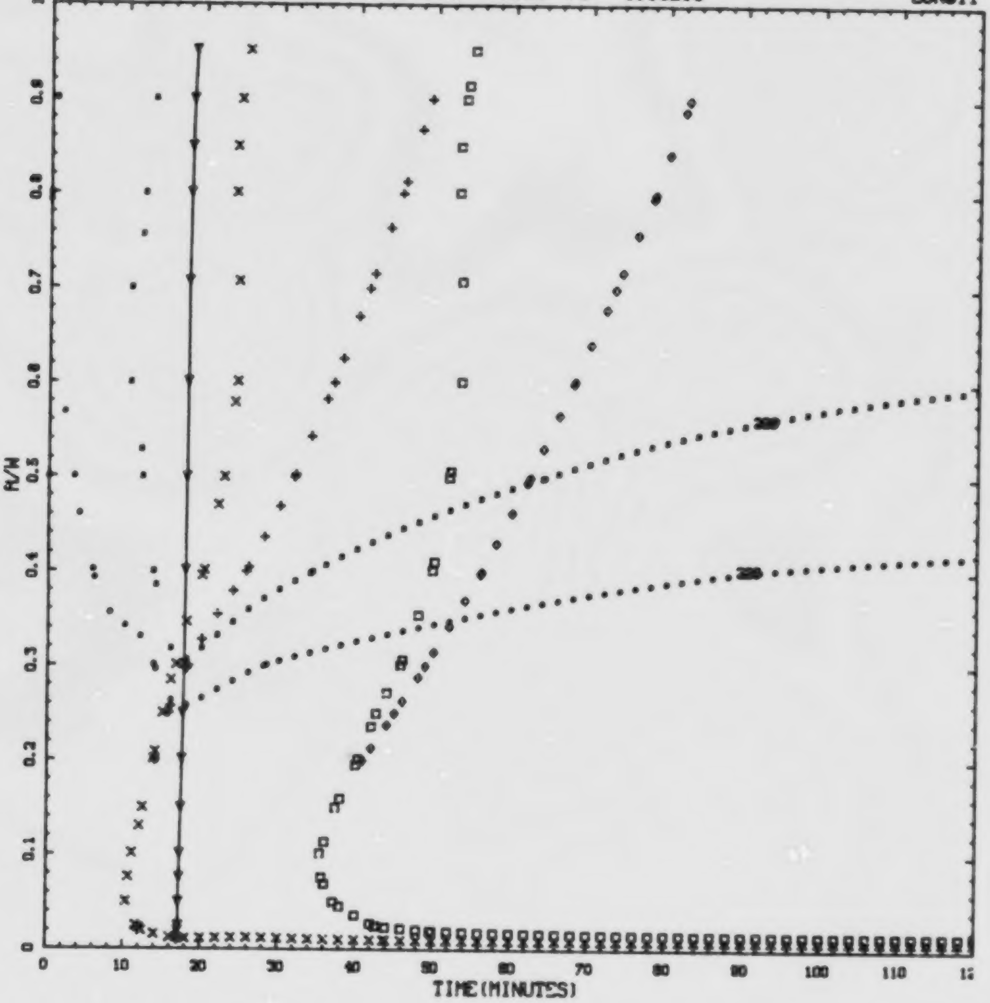


653

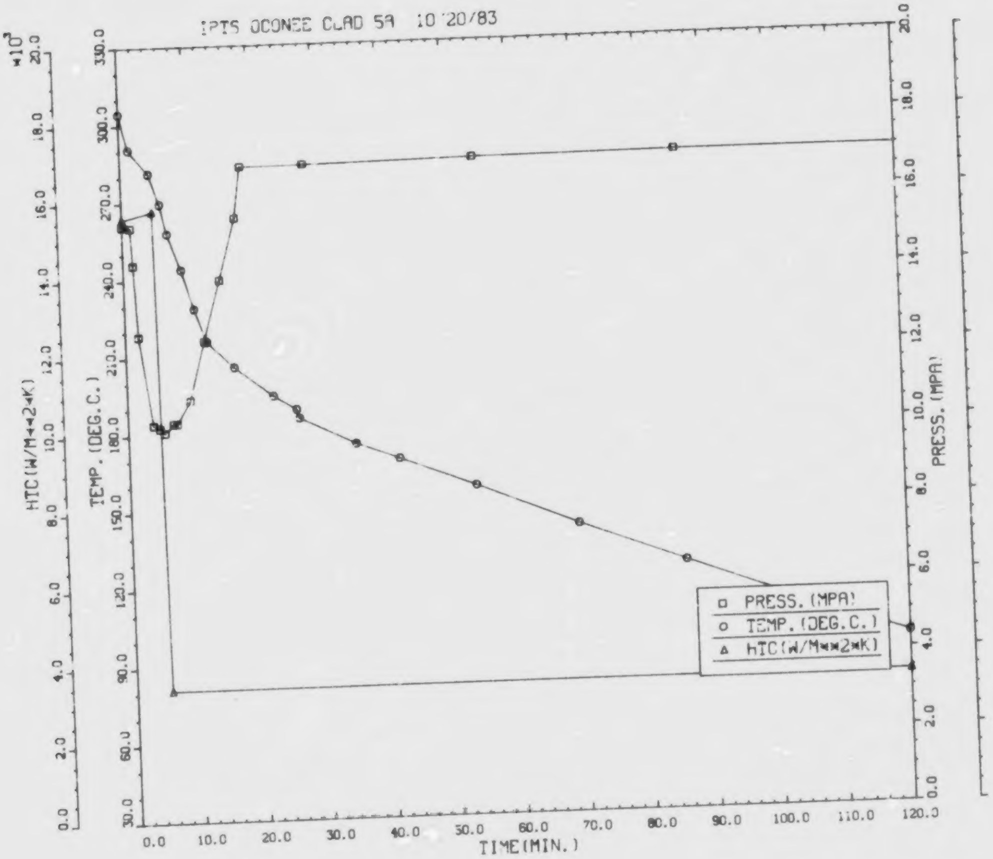

IPTS OCONEE CLAD W2TBV 11/04/83
TEMP. DEGC.
75
100
125
150
175
200
225
250
275
300
325
TIME
0
10
20
30
40
50
60
70
80
90
100
110
12

654

CRITICAL CRACK DEPTH CURVES FOR IPTS OCONEE CLAD W2TBV 11/04/83
RTNDT0 --6.7 DEGC %CU - 0.29 %NI - 0.55 F0 - 1.09E19 LONGIT



TRANSIENT NO. 41 [5A(LANL7)]


IPTS OCONEE CLAD 59 10/20/83
HTC(W/M**2*K) *10³
0.0
2.0
4.0
6.0
8.0
10.0
12.0
14.0
16.0
18.0
20.0
TEMP.(DEG.C.)
30.0
60.0
90.0
120.0
150.0
180.0
210.0
240.0
270.0
300.0
330.0
PRESS.(MPA)
0.0
2.0
4.0
6.0
8.0
10.0
12.0
14.0
16.0
18.0
20.0
TIME(MIN.)
0.0
10.0
20.0
30.0
40.0
50.0
60.0
70.0
80.0
90.0
100.0
110.0
120.0
□ PRESS.(MPA)
○ TEMP.(DEG.C.)
△ HTC(W/M**2*K)

657

```
IPTS OCONEE CLAD 5A  10/20/83                         1. FLAWS/M**3     FO =  1.090D+19

          -------------UNADJUSTED--------------           ---ADJUSTED----
WELD      P(F/E)      95%CI     %ERR   P(INITIA)    N*V     P(F/E)    %ERR     NTRIALS


 1        2.39D-02   2.27D-03    9.52   2.41D-02  0.040   9.55D-04                10000
 2        1.44D-02   1.26D-03    8.75   1.44D-02  0.040   5.74D-04                20000
 3        2.39D-02   2.27D-03    9.52   2.43D-02  0.010   2.39D-04                10000

                                              VESSEL      1.77D-03   6.01



 DEPTHS FOR INITIAL INITIATION (MM)
                   3.2   6.3  12.7  19.0  25.4  38.1  50.8
   NUMBER          333   653   235    70    21     2     0
   PERCENT        25.3  49.7  17.9   5.3   1.6   0.2   0.0



TIMES OF FAILURE(MINUTES)
                  0.0  10.0  20.0  30.0  40.0  50.0  60.0  70.0  80.0  90.0 100.0 110.0 120.0
   NUMBER               0     0     0     0     0     1     3    14    50   135   342   758
   PERCENT            0.0   0.0   0.0   0.0   0.0   0.1   0.2   1.1   3.8  10.4  26.2  58.2



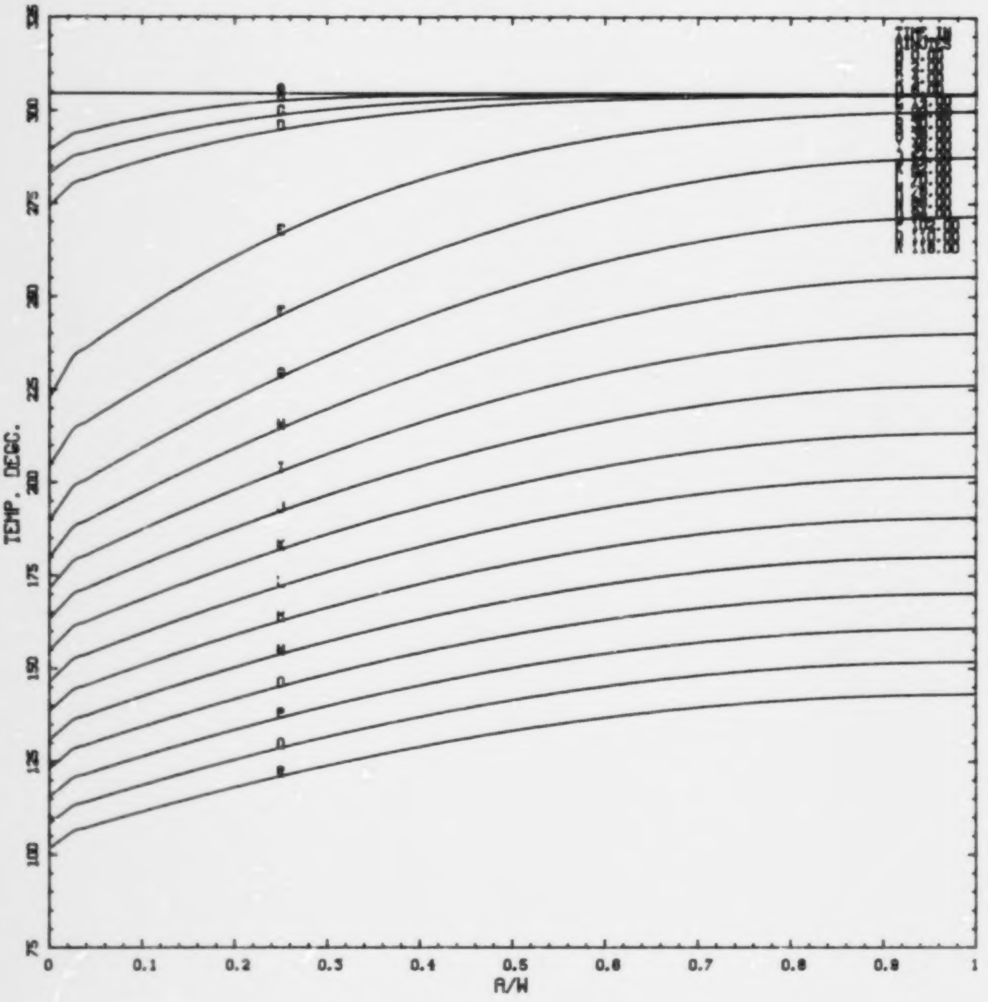
 INITIATION T-RTNDT(DEG.C)
                -55.6 -41.7 -27.8 -13.9   0.0  13.9  27.8  41.7  55.6  69.4  83.3  97.2 111.1
   NUMBER              13    76   293   514   363    56     1     0     0     0     0     0
   PERCENT            1.0   5.8  22.3  39.1  27.6   4.3   0.1   0.0   0.0   0.0   0.0   0.0



 ARREST T-RTNDT(DEG.C)
                -27.8 -13.9   0.0  13.9  27.8  41.7  55.6  69.4  83.3  97.2 111.1 125.0 138.9
   NUMBER               0     0     5     7     1     0     0     0     0     0     0     0
   PERCENT            0.0   0.0  38.5  53.8   7.7   0.0   0.0   0.0   0.0   0.0   0.0   0.0
```

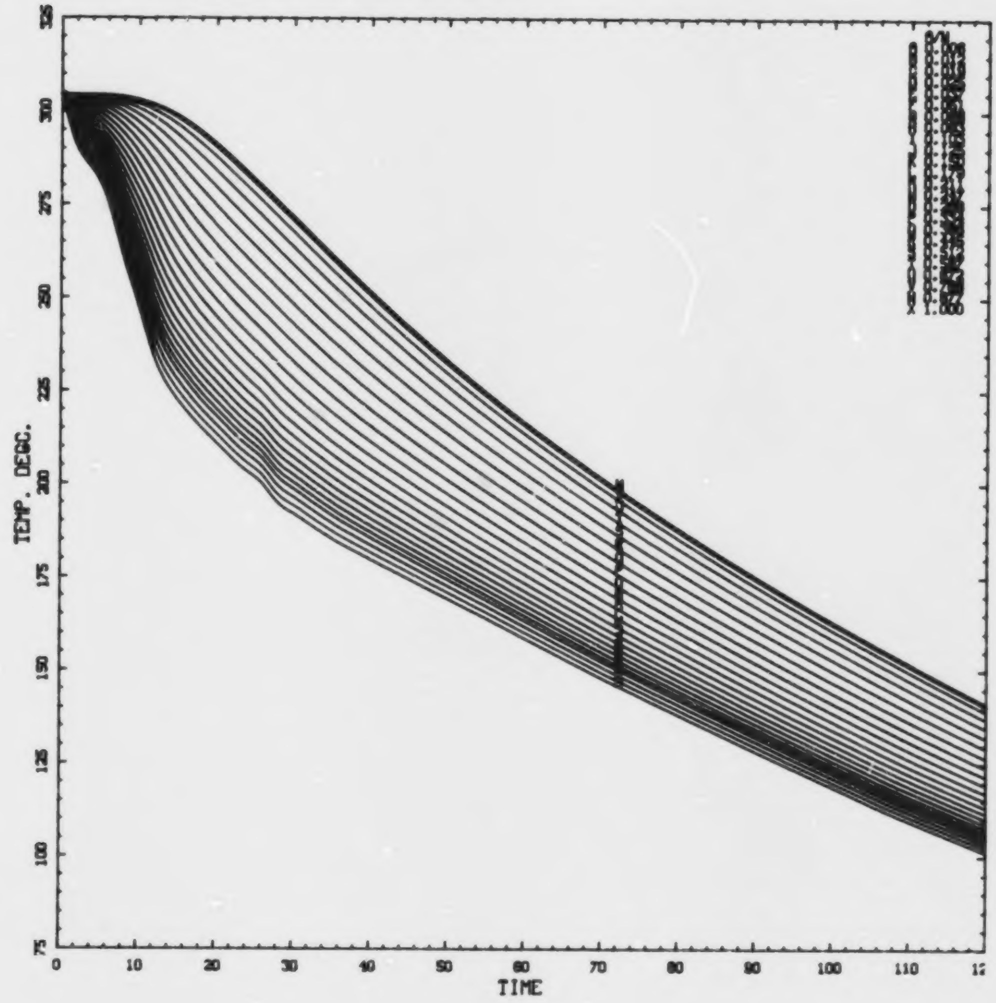
658

IPTS OCONEE CLAD 5A 10/20/83

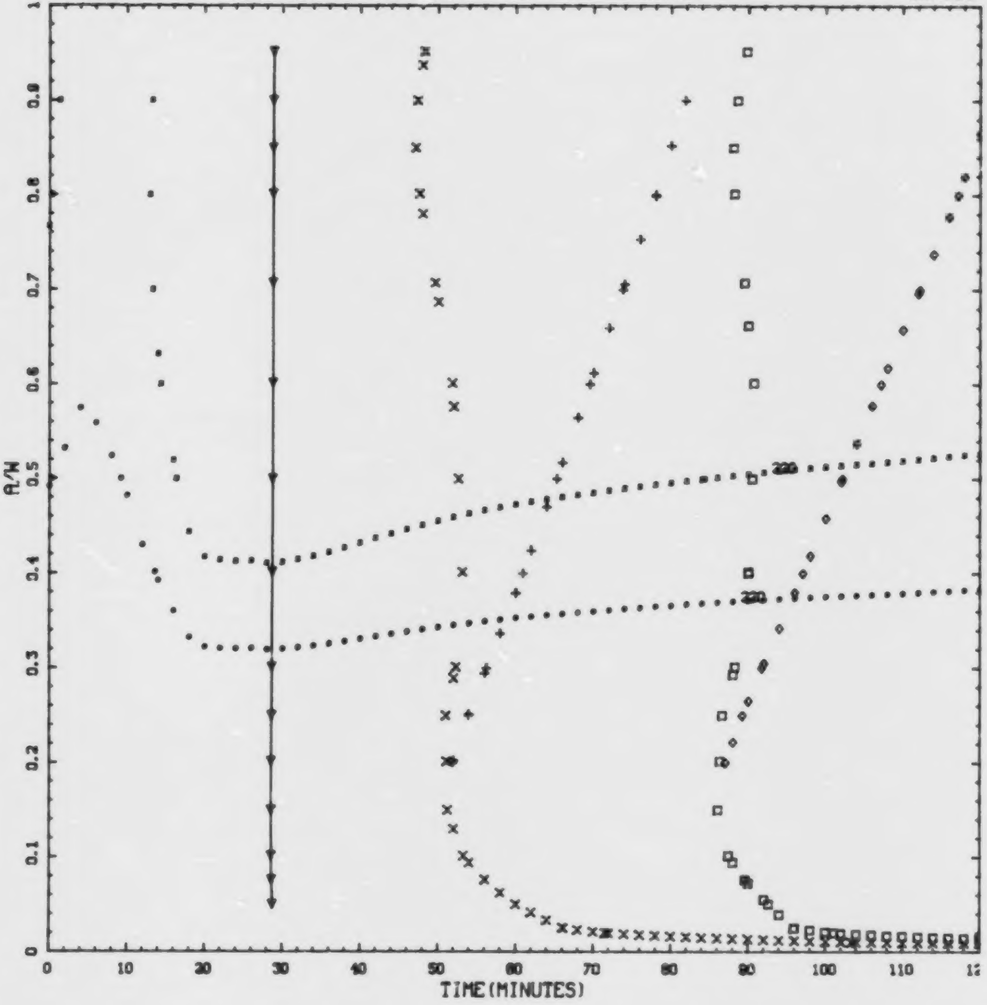


659

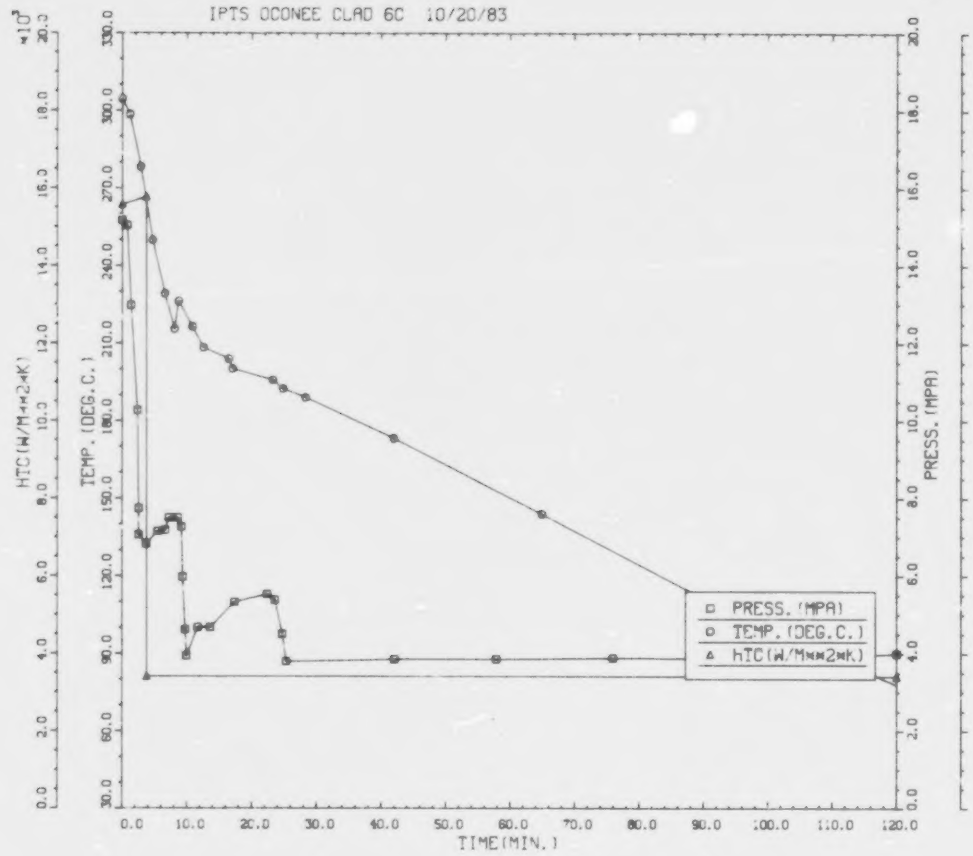
IPTS OCONEE CLAD 5A 10/20/83



660



TRANSIENT NO. 46 [6C(LANL12)]


IPTS OCONEE CLAD 6C 10/20/83
×10^3
HTC(W/M**2*K)
0.0
2.0
4.0
6.0
8.0
10.0
12.0
14.0
16.0
18.0
20.0
TEMP.(DEG.C.)
30.0
60.0
90.0
120.0
150.0
180.0
210.0
240.0
270.0
300.0
330.0
PRESS.(MPA)
TIME(MIN.)
0.0
10.0
20.0
30.0
40.0
50.0
60.0
70.0
80.0
90.0
100.0
110.0
120.0
□ PRESS.(MPA)
○ TEMP.(DEG.C.)
△ HTC(W/M**2*K)

```
IPTS OCONEE CLAD 6C  10/20/83                               1. FLAWS/M**3     F0 =  1.090D+19

        -------------UNADJUSTED--------------             ---ADJUSTED----
WELD    P(F/E)      95%CI     %ERR   P(INITIA)    N*V      P(F/E)    %ERR    NTRIALS


 1      1.41D-05   7.98D-06  56.58   5.62D-03  0.040   5.64D-07              500000
 2      8.22D-06   6.09D-06  74.08   3.81D-03  0.040   3.29D-07              500000
 3      1.64D-05   8.61D-06  52.38   6.11D-03  0.010   1.64D-07              500000

                                              VESSEL     1.06D-06  38.83


 DEPTHS FOR INITIAL INITIATION (MM)
               3.2   6.3  12.7  19.0  25.4  38.1  50.8
  NUMBER      3113  9042   683   292    86    15     2
  PERCENT     23.5  68.3   5.2   2.2   0.6   0.1   0.0



TIMES OF FAILURE(MINUTES)
               0.0  10.0  20.0  30.0  40.0  50.0  60.0  70.0  80.0  90.0 100.0 110.0 120.0
  NUMBER             0     0     0     0     0     0     0     0     0     0     0    33
  PERCENT          0.0   0.0   0.0   0.0   0.0   0.0   0.0   0.0   0.0   0.0   0.0 100.0

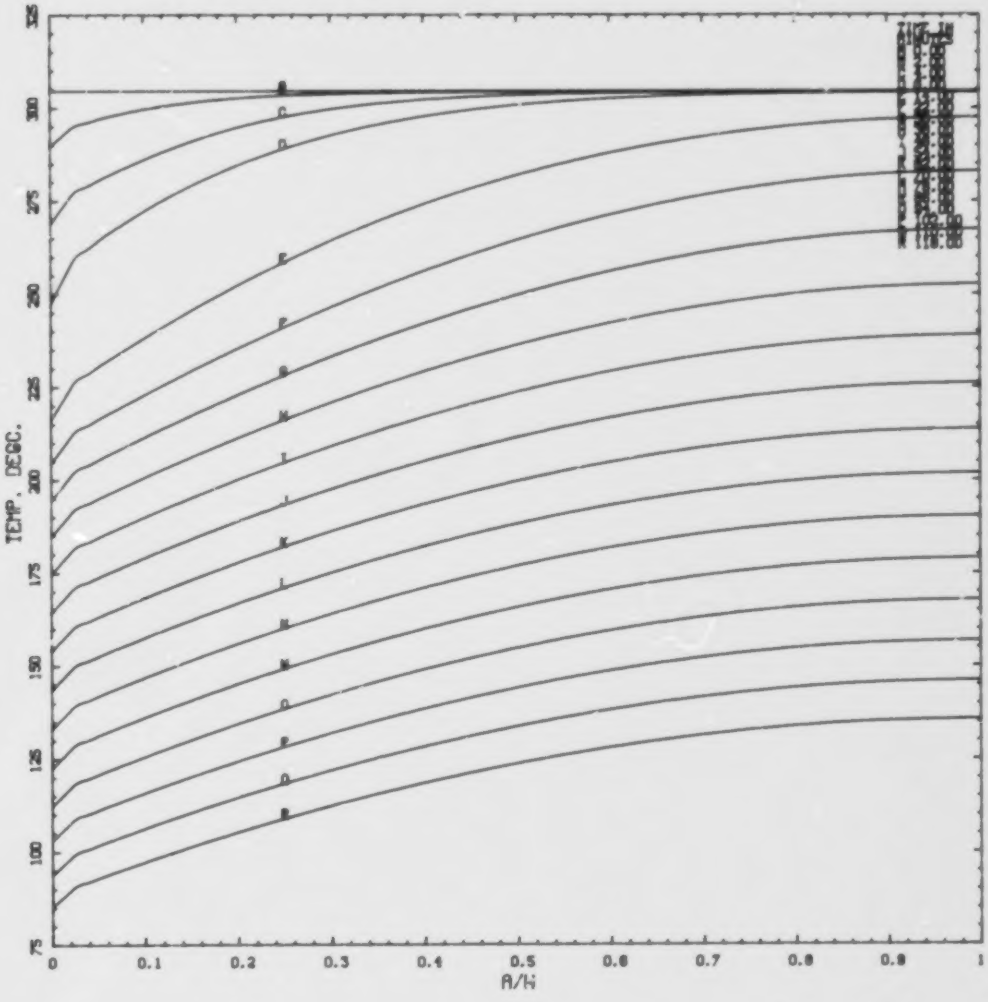


 INITIATION T-RTNDT(DEG.C)
             -55.6 -41.7 -27.8 -13.9   0.0  13.9  27.8  41.7  55.6  69.4  83.3  97.2 111.1
  NUMBER            1250  3605  5619  2882   342    66     8     0     0     0     0     0
  PERCENT            9.1  26.2  40.8  20.9   2.5   0.5   0.1   0.0   0.0   0.0   0.0   0.0

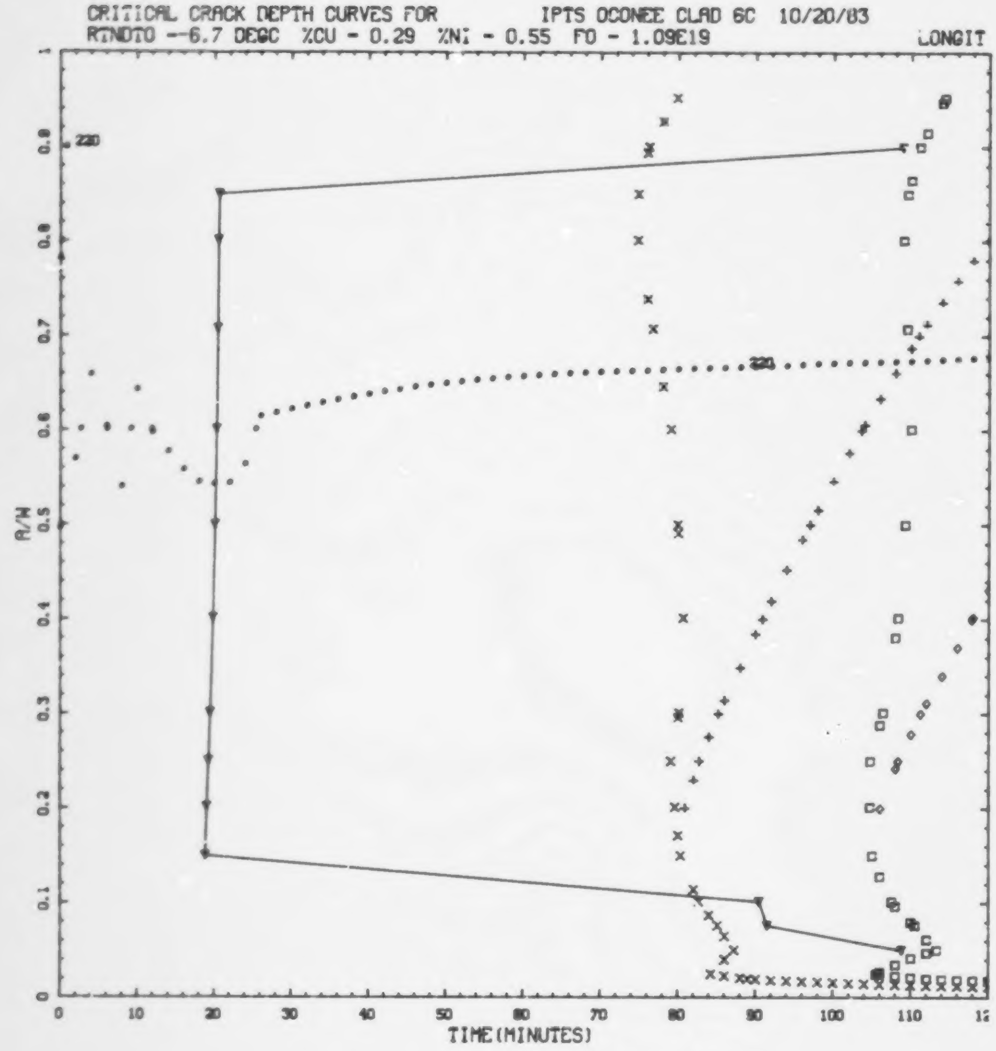


 ARREST T-RTNDT(DEG.C)
             -27.8 -13.9   0.0  13.9  27.8  41.7  55.6  69.4  83.3  97.2 111.1 125.0 138.9
  NUMBER            3366  3879   991   172  2516  1898     0     0     0     0     0     0
  PERCENT           26.3  30.3   7.7   1.3  19.6  14.8   0.0   0.0   0.0   0.0   0.0   0.0
```

IPTS OCONEE CLAD 6C 10/20/83



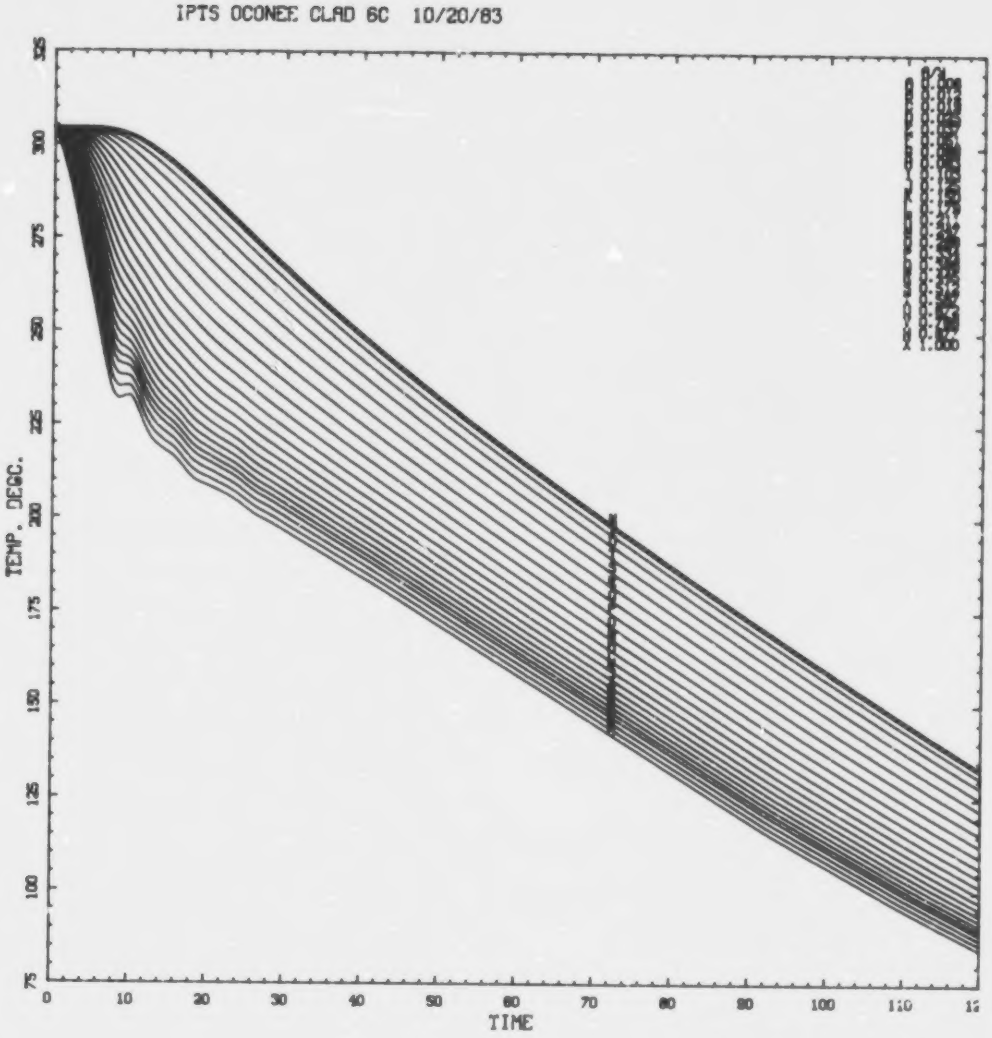
465



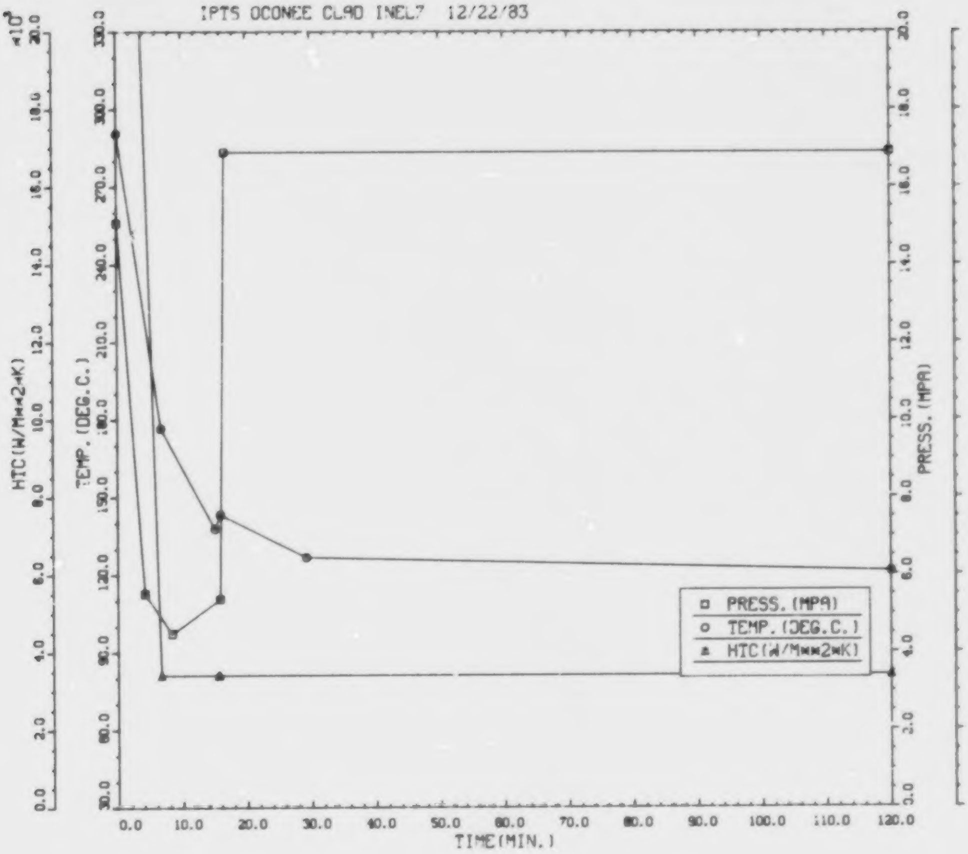
CRITICAL CRACK DEPTH CURVES FOR IPTS OCONEE CLAD 6C 10/20/83
RTNDTO --6.7 DEGC %CU - 0.29 %NI - 0.55 FO - 1.09E19 LONGIT

666

IPTS OCONEE CLAD 6C 10/20/83



TRANSIENT NO. 57 (INEL7)


IPTS OCONEE CLAD INEL7 12/22/83
HTC(W/M**2*K) *10³
0.0
2.0
4.0
6.0
8.0
10.0
12.0
14.0
16.0
18.0
20.0
TEMP.(DEG.C.)
30.0
60.0
90.0
120.0
150.0
180.0
210.0
240.0
270.0
300.0
330.0
PRESS.(MPA)
0.0
2.0
4.0
6.0
8.0
10.0
12.0
14.0
16.0
18.0
20.0
TIME(MIN.)
0.0
10.0
20.0
30.0
40.0
50.0
60.0
70.0
80.0
90.0
100.0
110.0
120.0
□ PRESS.(MPA)
○ TEMP.(DEG.C.)
▲ HTC(W/M**2*K)

```
IPTS OCONEE CLAD INEL7  12/22/83                          1. FLAWS/M**3     F0 =  1.090D+19

        -------------UNADJUSTED--------------            ---ADJUSTED----
WELD    P(F/E)      95%CI     %ERR   P(INITIA)   N*V     P(F/E)    %ERR     NTRIALS


 1      2.26D-03   2.15D-04   9.50   2.29D-03  0.040   9.06D-05              110000
 2      9.81D-04   9.60D-05   9.79   9.81D-04  0.040   3.93D-05              240000
 3      2.17D-03   2.10D-04   9.71   2.18D-03  0.010   2.17D-05              110000

      .                                      VESSEL    1.51D-04   6.37



 DEPTHS FOR INITIAL INITIATION (MM)
               3.2   6.3  12.7  19.0  25.4  38.1  50.8
   NUMBER      346   536   275    58    18     4     1
   PERCENT    27.9  43.3  22.2   4.7   1.5   0.3   0.1




TIMES OF FAILURE(MINUTES)
                0.0  10.0  20.0  30.0  40.0  50.0  60.0  70.0  80.0  90.0 100.0 110.0 120.0
   NUMBER          0    93   727   357     8    10    11     7     6     8     3     1
   PERCENT       0.0   7.6  59.1  29.0   0.6   0.8   0.9   0.6   0.5   0.6   0.2   0.1

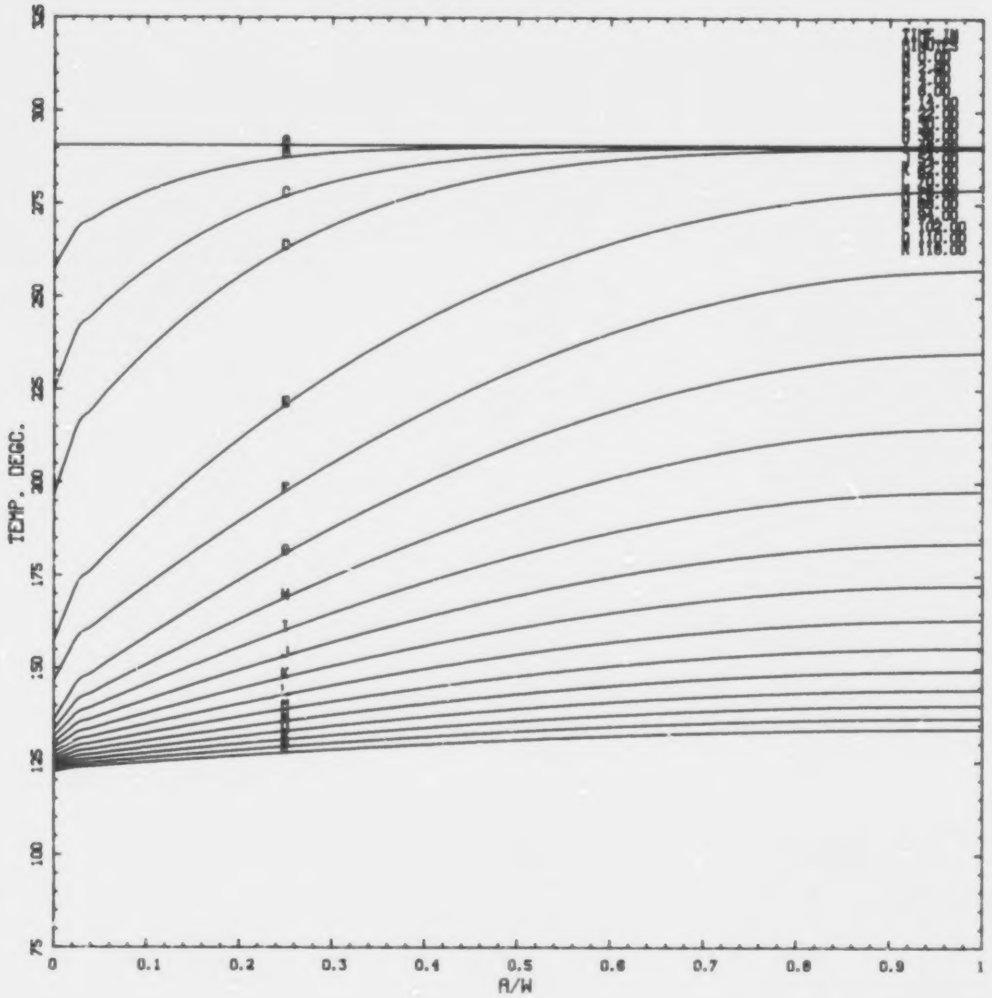



 INITIATION T-RTNDT(DEG.C)
              -55.6 -41.7 -27.8 -13.9   0.0  13.9  27.8  41.7  55.6  69.4  83.3  97.2 111.1
   NUMBER          0     0    23   184   526   446    99     8     0     0     0     0
   PERCENT       0.0   0.0   1.8  14.3  40.9  34.7   7.7   0.6   0.0   0.0   0.0   0.0

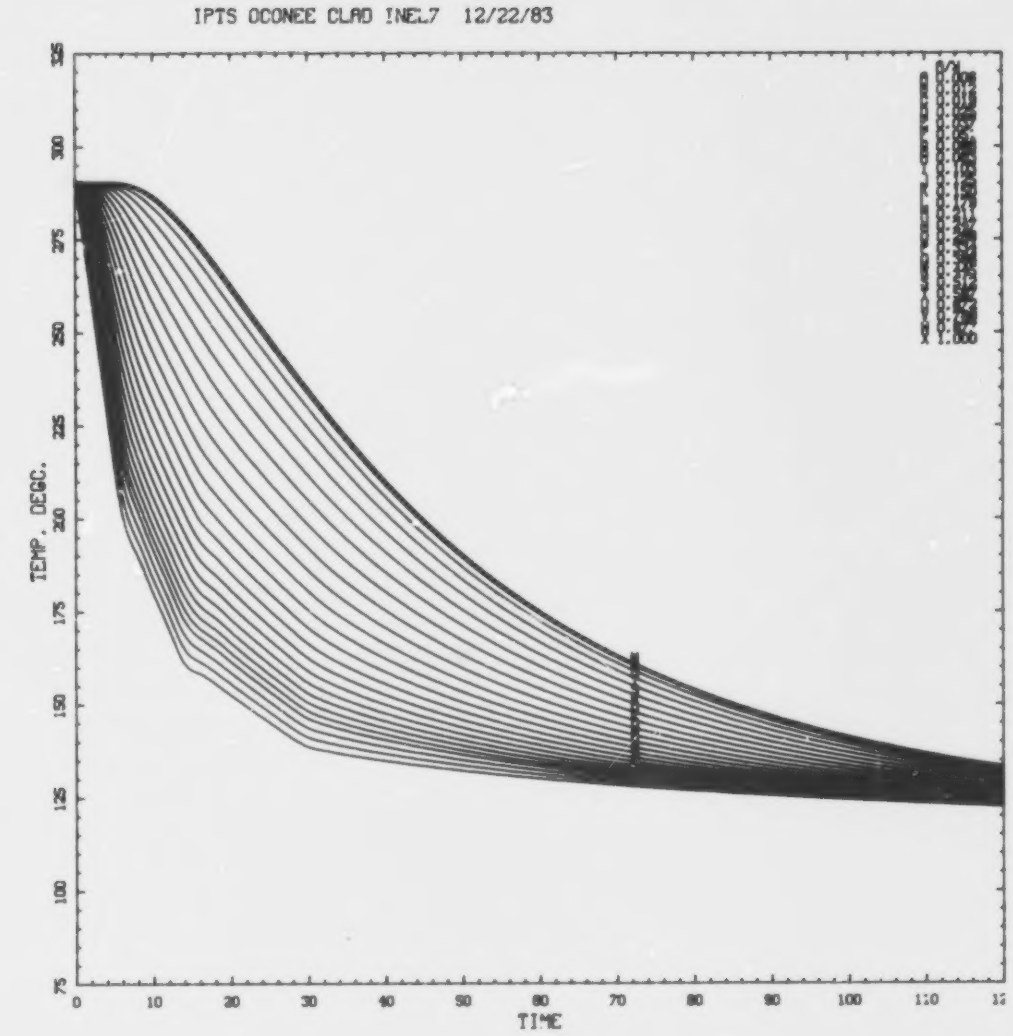



 ARREST T-RTNDT(DEG.C)
              -27.8 -13.9   0.0  13.9  27.8  41.7  55.6  69.4  83.3  97.2 111.1 125.0 138.9
   NUMBER          0     0     0     0     0     0     0    21    34     0     0     0
   PERCENT       0.0   0.0   0.0   0.0   0.0   0.0   0.0  38.2  61.8   0.0   0.0   0.0
```

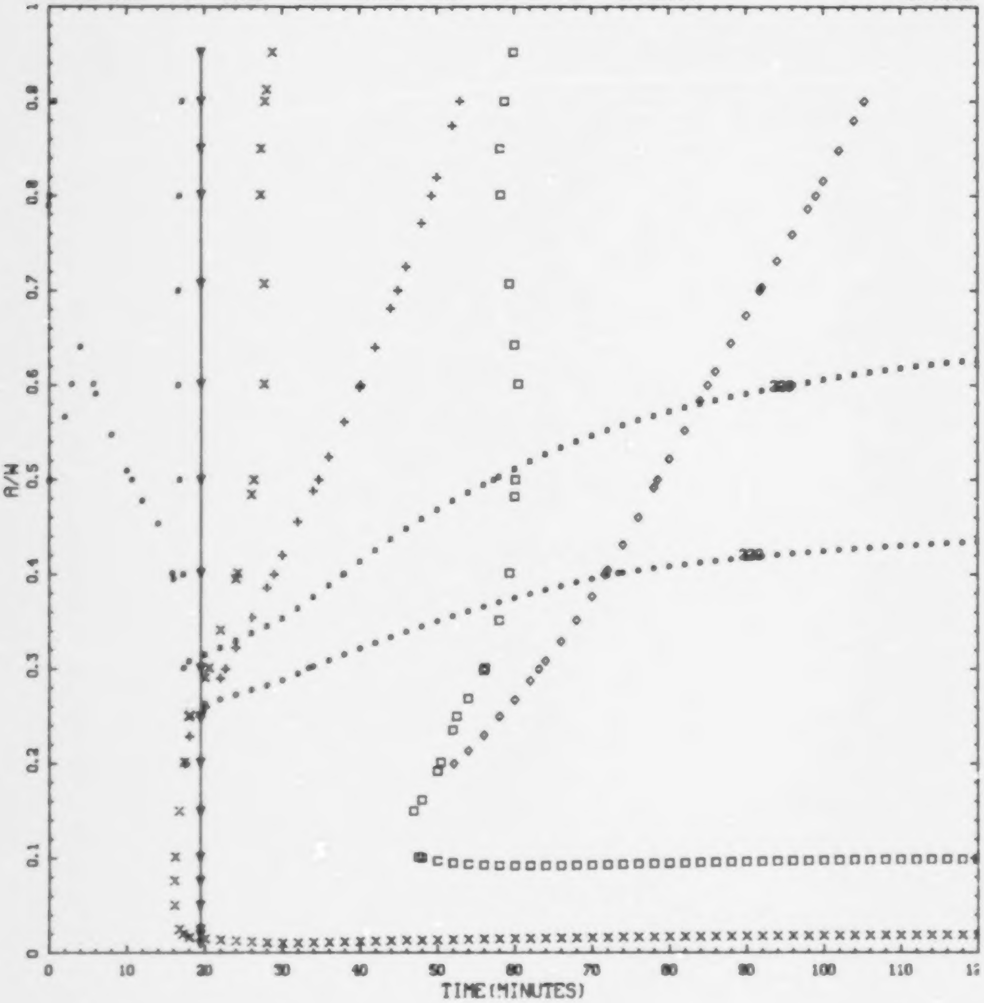
IPTS OCONEE CLAD INEL7 12/22/83



671


IPTS OCONEE CLAD INEL7 12/22/83
TEMP. DEGC.
75
100
125
150
175
200
225
250
275
300
325
TIME
0
10
20
30
40
50
60
70
80
90
100
110
12
1.000

CRITICAL CRACK DEPTH CURVES FOR IPTS OCONEE CLAD INEL7 12/22/83
RTNDTO --6.7 DEGC %CU - 0.29 %NI - 0.55 FO - 1.09E19 LONGIT



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APPENDIX F

ALTERNATIVE APPROACH TO SEQUENCE ASSIGNMENT

A review of the assignment of representative fracture-mechanics cases to each sequence, as presented in Table A.3 of Appendix A, revealed both conservative and non-conservative assignments. It was the opinion of the analysts that in general conservative and sometimes overly conservative assignments were made. This was particularly true for the assignment of a fracture mechanics case to the residual cases. An attempt was made to reconstruct the table and thus reduce the level of conservatism involved in this portion of the Oconee analysis. Unfortunately, there was little that could be done without performing additional thermal-hydraulic and/or fracture-mechanics calculations. In addition the resolution of the few non-conservative assignments led to a higher overall vessel failure value. Thus in trying to reduce both the conservative and non-conservative biases introduced by the assignment of fracture-mechanics cases, we produced a set of data which was clearly biased in the conservative direction. This data is presented here as a separate appendix since it includes the elimination of some of the non-conservative biases found in the analysis even though it fails to address the larger category of conservative biases.

REC	TYPE	INIT	SPC	FWS	ECC	PPC	EFREQ	TRANS	HF	TFREQ	P(F|T)	P(TWC)
1	NORM	RT	0	0	0	0	5.0E+00	NONE	1.0E+00	5.0E+00	0.0E+00	0.0E+00
2	FW	RT	0	1	0	0	1.6E-01	OVRFD3	1.0E+00	1.6E-01	1.0E-07	1.6E-08
3	FW	RT	0	3	0	1	4.9E-04	PSBG2	9.9E-01	4.9E-04	1.0E-07	4.9E-11
4	FW	RT	0	3	0	1	4.9E-04	PSBG4	1.0E-02	4.9E-06	1.3E-03	6.4E-09
5	FW	RT	0	3	0	2	3.0E-05	PSBG2	9.9E-01	3.0E-05	1.0E-07	3.0E-12
6	FW	RT	0	3	0	2	3.0E-05	PSBG4	1.0E-02	3.0E-07	1.3E-03	3.9E-10
7	FW	RT	0	4	0	0	1.9E-03	NONE	9.9E-01	1.9E-03	0.0	0.0
8	FW	RT	0	4	0	0	1.9E-03	OVRFD6	1.0E-02	1.9E-05	1.4E-06	2.7E-11
9	FW	RT	0	4	0	0	1.9E-03	OVRFD7	3.6E-03	6.8E-06	1.0E-07	6.8E-13
10	FW	RT	0	4	0	1	4.7E-05	NONE	9.9E-01	4.7E-05	0.0	0.0
11	FW	RT	0	4	0	1	4.7E-05	PSBG4	1.0E-02	4.7E-07	1.3E-03	6.1E-10
12	FW	RT	0	4	0	1	4.7E-05	PSBG2	3.6E-03	1.7E-07	1.0E-07	1.7E-14
13	FW	RT	0	4	0	2	1.3E-04	NONE	9.9E-01	1.3E-04	0.0	0.0
14	FW	RT	0	4	0	2	1.3E-04	PSBG4	1.0E-02	1.3E-06	1.3E-03	1.7E-09
15	FW	RT	0	4	0	2	1.3E-04	PSBG2	3.6E-03	4.7E-07	1.0E-07	4.7E-14
16	FW	RT	0	2	0	0	5.8E-04	MIN	9.9E-01	5.7E-04	1.0E-07	5.7E-11
17	FW	RT	0	2	0	0	5.8E-04	INEL6	1.0E-02	5.8E-06	4.2E-07	2.4E-12
18	FW	RT	0	2	0	1	1.4E-05	PSBG2	9.9E-01	1.4E-05	1.0E-07	1.4E-12
19	FW	RT	0	2	0	1	1.4E-05	INEL6	1.0E-02	1.4E-07	4.2E-07	5.9E-14
20	FW	RT	0	2	0	2	3.7E-05	PSBG2	9.9E-01	3.7E-05	1.0E-07	3.7E-12
21	FW	RT	0	2	0	2	3.7E-05	INEL6	1.0E-02	3.7E-07	4.2E-07	1.6E-13
22	SL	RT	1	0	0	0	6.9E-01	TBVG2	9.9E-01	6.8E-01	1.0E-07	6.8E-08
23	SL	RT	1	0	0	0	6.9E-01	TBVG1	1.0E-02	6.9E-03	4.0E-07	2.8E-09
24	SL	RT	1	0	0	1	1.7E-02	PSBG5	9.9E-01	1.7E-02	1.0E-07	1.7E-09
25	SL	RT	1	0	0	1	1.7E-02	TBVG6	1.0E-02	1.7E-04	1.7E-06	2.9E-10
26	SL	RT	1	0	0	2	4.5E-02	PSBG5	9.9E-01	4.5E-02	1.0E-07	4.5E-09
27	SL	RT	1	0	0	2	4.5E-02	TBVG6	1.0E-02	4.5E-04	1.7E-06	7.7E-10
28	SL	RT	1	0	1	0	9.0E-04	TBVG2	9.9E-01	8.9E-04	1.0E-07	8.9E-11
29	SL	RT	1	0	1	0	9.0E-04	TBVG1	1.0E-02	9.0E-06	4.0E-07	3.6E-12
30	SL	RT	1	0	1	1	2.2E-05	PSBG5	9.9E-01	2.2E-05	1.0E-07	2.2E-12
31	SL	RT	1	0	1	1	2.2E-05	TBVG6	1.0E-02	2.2E-07	1.7E-06	3.7E-13
32	SL	RT	1	0	1	2	5.6E-05	PSBG5	9.9E-01	5.5E-05	1.0E-07	5.5E-12
33	SL	RT	1	0	1	2	5.6E-05	TBVG6	1.0E-02	5.6E-07	1.7E-06	9.5E-13
34	SS	RT	1	3	0	1	2.8E-05	PSBG5	9.9E-01	2.8E-05	1.0E-07	2.8E-12
35	SS	RT	1	3	0	1	2.8E-05	TBVG6	1.0E=02	2.8E-07	1.7E-06	4.8E-13
36	SS	RT	1	3	0	2	7.7E-05	PSBG5	9.9E-01	7.6E-05	1.0E-07	7.6E-12
37	SS	RT	1	3	0	2	7.7E-05	TBVG6	1.0E-02	7.7E-07	1.7E-06	1.3E-12
38	SS	RT	1	4	0	0	5.0E-03	TBVG2	9.9E-01	5.0E-03	1.0E-07	5.0E-10
39	SS	RT	1	4	0	0	5.0E-03	TBVG1	1.0E-02	5.0E-05	4.0E-07	2.0E-11
40	SS	RT	1	4	0	0	5.0E-03	TBVG1	3.6E-03	1.8E-06	4.0E-07	7.2E-13
41	SS	RT	1	4	0	1	1.2E-04	PSBG5	9.9E-01	1.2E-04	1.0E-07	1.2E-11
42	SS	RT	1	4	0	1	1.2E-04	TBVG6	1.0E-02	1.2E-06	1.7E-06	2.0E-12
43	SS	RT	1	4	0	1	1.2E-04	TBVG6	3.6E-03	4.3E-07	1.7E-06	7.3E-13
44	SS	RT	1	4	0	2	3.2E-04	PSBG5	9.9E-01	3.2E-04	1.0E-07	3.2E-11
45	SS	RT	1	4	0	2	3.2E-04	TBVG6	1.0E-02	3.2E-06	1.7E-06	5.4E-12
46	SS	RT	1	4	0	2	3.2E-04	TBVG6	3.6E-03	1.2E-06	1.7E-06	2.0E-12
47	SS	RT	1	4	1	0	4.4E-06	TBVG2	9.9E-01	4.4E-06	1.0E-07	4.4E-13
48	SS	RT	1	4	1	0	4.4E-06	TBVG1	1.0E-02	4.4E-08	4.0E-07	1.8E-14
49	SS	RT	1	4	1	0	4.4E-06	TBVG1	3.6E-03	1.6E-08	4.0E-07	6.4E-15
50	SL	RT	1	1	0	0	2.2E-02	TBVG3	9.9E-01	2.2E-02	1.0E-07	2.2E-0[illegible]
51	SL	RT	1	1	0	0	2.2E-02	TBVG4	1.0E-02	2.2E-04	2.0E-03	4.4E-07
52	SS	RT	1	1	0	1	5.4E-04	TBVG6	9.9E-01	5.3E-04	1.7E-06	9.0E-10
53	SS	RT	1	1	0	1	5.4E-04	TBVG6	1.0E-02	5.4E-06	1.7E-06	9.2E-12

REC	TYPE	INIT	SPC	FWS	ECC	PPC	EFREQ	TRANS	HF	TFREQ	P(F|T)	P(TWC)
54	SS	RT	1	1	0	2	1.4E-03	TBVG6	9.9E-01	1.4E-03	1.7E-06	2.4E-09
55	SS	RT	1	1	0	2	1.4E-03	TBVG6	1.0E-02	1.4E-05	1.7E-06	2.4E-11
56	SS	RT	1	1	1	0	2.8E-05	TBVG3	9.9E-01	2.8E-05	1.0E-07	2.8E-12
57	SS	RT	1	1	1	0	2.8E-05	TBVG4	1.0E-02	2.8E-07	2.0E-03	5.6E-10
58	SS	RT	1	2	0	0	8.8E-05	TBVG3	9.9E-01	8.7E-05	1.0E-07	8.7E-12
59	SS	RT	1	2	0	0	8.8E-05	TBVG4	1.0E-02	8.8E-07	2.0E-03	1.9E-09
60	SS	RT	1	2	0	1	2.2E-06	TBVG4	9.9E-01	2.2E-06	2.0E-03	4.4E-09
61	SS	RT	1	2	0	1	2.2E-06	TBVG4	1.0E-02	2.2E-08	2.0E-03	4.4E-11
62	SS	RT	1	2	0	2	5.0E-06	TBVG4	9.9E-01	5.0E-06	2.0E-03	1.0E-08
63	SS	RT	1	2	0	2	5.0E-06	TBVG4	1.0E-02	5.0E-08	2.0E-03	1.0E-10
64	SL	RT	2	0	0	0	8.2E-03	TBVG2	9.9E-01	8.[illegible]E-03	1.0E-07	8.1E-10
65	SL	RT	2	0	0	0	8.2E-03	TBVG1	1.0E-02	8.2E-05	4.0E-07	3.3E-11
66	SL	RT	2	0	0	1	2.0E-04	PSBG5	9.9E-01	2.0E-04	1.0E-07	2.0E-11
67	SL	RT	2	0	0	1	2.0E-04	TBVG6	1.0E-02	2.0E-06	1.7E-06	3.4E-12
68	SL	RT	2	0	0	2	5.3E-04	PSBG5	9.9E-01	5.2E-04	1.0E-07	5.2E-11
69	SL	RT	2	0	0	2	5.3E-04	TBVG6	1.0E-02	5.3E-06	1.7E-06	9.0E-12
70	SL	RT	2	0	1	0	9.2E-06	TBVG2	9.9E-01	9.1E-06	1.0E-07	9.1E-13
71	SL	RT	2	0	1	0	9.2E-06	TBVG1	1.0E-02	9.2E-08	4.0E-07	3.7E-14
72	SS	RT	2	3	0	0	1.2E-05	TBVG2	9.9E-01	1.2E-05	1.0E-07	1.2E-12
73	SS	RT	2	3	0	0	1.2E-05	LANL7	1.0E-02	1.2E-07	1.8E-03	2.2E-10
74	SS	RT	2	4	0	0	5.5E-05	TBVG2	9.9E-01	5.4E-05	1.0E-07	5.4E-12
75	SS	RT	2	4	0	0	5.5E-05	OVRFD6	1.0E-02	5.5E-07	1.4E-06	7.7E-13
76	SS	RT	2	4	0	0	5.5E-05	OVRFD7	3.6E-03	2.0E-07	1.0E-07	2.0E-14
77	SS	RT	2	4	0	2	2.6E-06	TBVG10	9.9E-01	2.6E-06	1.0E-05	2.6E-11
78	SS	RT	2	4	0	2	2.6E-06	TBVG10	1.0E-02	2.6E-08	1.0E-05	2.6E-13
79	SS	RT	2	4	0	2	2.6E-06	TBVG6	3.6E-03	9.4E-08	1.7E-06	1.6E-13
80	SS	RT	2	1	0	0	2.6E-04	TBVG3	9.9E-01	2.6E-04	1.0E-07	2.6E-11
81	SS	RT	2	1	0	0	2.6E-04	TBVG4	1.0E-02	2.6E-06	2.0E-03	5.2E-09
82	SS	RT	2	1	0	1	5.8E-06	TBVG6	9.9E-01	5.7E-06	1.7E-06	9.7E-12
83	SS	RT	2	1	0	1	5.8E-06	TBVG6	1.0E-02	5.8E-08	1.7E-06	9.9E-14
84	SS	RT	2	1	0	2	1.3E-05	TBVG6	9.9E-01	1.3E-05	1.7E-06	2.2E-11
85	SS	RT	2	1	0	2	1.3E-05	TBVG6	1.0E-02	1.3E-07	1.7E-06	2.2E-13
86	SL	RT	3	0	0	0	7.6E-02	TBVG5	9.9E-01	7.5E-02	5.2E-06	3.9E-07
87	SL	RT	3	0	0	0	7.6E-02	TBVG5	1.0E-02	7.6E-04	5.2E-06	4.0E-09
88	SL	RT	3	0	0	1	1.9E-03	TBVG10	9.9E-01	1.9E-03	1.0E-05	1.9E-08
89	SL	RT	3	0	0	1	1.9E-03	TBVG10	1.0E-02	1.9E-05	1.0E-05	1.9E-10
90	SL	RT	3	0	0	2	5.0E-03	TBVG10	9.9E-01	5.0E-03	1.0E-05	5.0E-08
91	SL	RT	3	0	0	2	5.0E-03	TBVG10	1.0E-02	5.0E-05	1.0E-05	5.0E-10
92	SL	RT	3	0	1	0	9.9E-05	TBVG5	9.9E-01	9.8E-05	5.2E-06	5.1E-10
93	SL	RT	3	0	1	0	9.9E-05	TBVG5	1.0E-02	9.9E-07	5.2E-06	5.1E-12
94	SL	RT	3	0	1	1	2.4E-06	TBVG10	9.9E-01	2.4E-06	1.0E-05	2.4E-11
95	SL	RT	3	0	1	1	2.4E-06	TBVG10	1.0E-02	2.4E-08	1.0E-05	2.4E-13
96	SL	RT	3	0	1	2	5.5E-06	TBVG10	9.9E-01	5.4E-06	1.0E-05	5.4E-11
97	SL	RT	3	0	1	2	5.5E-06	TBVG10	1.0E-02	5.5E-08	1.0E-05	5.5E-13
98	SS	RT	3	3	0	0	1.4E-04	TBVG5	9.9E-01	1.4E-04	5.2E-06	7.3E-10
99	SS	RT	3	3	0	0	1.4E-04	LANL7	1.0E-02	1.4E-06	1.8E-03	2.5E-09
100	SS	RT	3	3	0	1	3.1E-06	TBVG10	9.9E-01	3.1E-06	1.0E-05	3.1E-11
101	SS	RT	3	3	0	1	3.1E-06	TBVG10	1.0E-02	3.1E-08	1.0E-05	3.1E-13
102	SS	RT	3	3	0	2	7.2E-06	TBVG10	9.9E-01	7.1E-06	1.0E-05	7.1E-11
103	SS	RT	3	3	0	2	7.2E-06	TBVG10	1.0E-02	7.1E-08	1.0E-05	7.1E-13
104	SS	RT	3	4	0	0	5.5E-04	TBVG5	9.9E-01	5.4E-04	5.2E-06	2.8E-09
105	SS	RT	3	4	0	0	5.5E-04	TBVG9	1.0E-02	5.5E-06	2.0E-03	1.1E-08
106	SS	RT	3	4	0	0	5.5E-04	TBVG9	3.6E-03	2.0E-06	2.0E-03	4.0E-09

REC	TYPE	INIT	SPC	FWS	ECC	PPC	EFREQ	TRANS	HF	TFREQ	P(F|T)	P(TWC)
107	SS	RT	3	4	0	1	1.3E-05	TBVG10	9.9E-01	1.3E-05	1.0E-05	1.3E-10
108	SS	RT	3	4	0	1	1.3E-05	TBVG10	1.0E-02	1.3E-07	1.0E-05	1.3E-12
109	SS	RT	3	4	0	1	1.3E-05	TBVG10	3.6E-03	4.7E-08	1.0E-05	4.7E-13
110	SS	RT	3	4	0	2	3.3E-05	TBVG10	9.9E-01	3.3E-05	1.0E-05	3.3E-10
111	SS	RT	3	4	0	2	3.3E-05	TBVG10	1.0E-02	3.3E-07	1.0E-05	3.3E-12
112	SS	RT	3	4	0	2	3.3E-05	TBVG10	3.6E-03	1.2E-07	1.0E-05	1.2E-12
113	SS	RT	3	1	0	0	2.4E-03	TBVG9	9.9E-01	2.4E-03	2.0E-03	4.8E-06
114	SS	RT	3	1	0	0	2.4E-03	TBVG9	1.0E-02	2.4E-05	2.0E-03	4.8E-08
115	SS	RT	3	1	0	1	6.0E-05	TBVG10	9.9E-01	5.9E-05	1.0E-05	5.9E-10
116	SS	RT	3	1	0	1	6.0E-05	TBVG10	1.0E-02	5.9E-07	1.0E-05	5.9E-12
117	SS	RT	3	1	0	2	1.6E-04	TBVG10	9.9E-01	1.6E-04	1.0E-05	1.6E-09
118	SS	RT	3	1	0	2	1.6E-04	TBVG10	1.0E-02	1.6E-06	1.0E-05	1.6E-11
119	SS	RT	3	1	1	0	2.6E-06	TBVG9	9.9E-01	2.6E-06	2.0E-03	5.2E-09
120	SS	RT	3	1	1	0	2.6E-06	TBVG9	1.0E-02	2.6E-08	2.0E-03	5.2E-11
121	SS	RT	3	2	0	0	8.7E-06	TBVG9	9.9E-01	8.6E-06	2.0E-03	1.7E-08
122	SS	RT	3	2	0	0	8.7E-06	TBVG9	1.0E-02	8.6E-08	2.0E-03	1.7E-10
123	SL	RT	4	0	0	0	6.9E-04	TBVG5	9.9E-01	6.8E-04	5.2E-06	3.5E-09
124	SL	RT	4	0	0	0	6.9E-04	TBVG5	1.0E-02	6.9E-06	5.2E-06	3.6E-11
125	SL	RT	4	0	0	1	1.5E-05	TBVG10	9.9E-01	1.5E-05	1.0E-05	1.5E-10
126	SL	RT	4	0	0	1	1.5E-05	TBVG10	1.0E-02	1.5E-07	1.0E-05	1.5E-12
127	SL	RT	4	0	0	2	4.3E-05	TBVG10	9.9E-01	4.3E-05	1.0E-05	4.3E-10
128	SL	RT	4	0	0	2	4.3E-05	TBVG10	1.0E-02	4.3E-07	1.0E-05	4.3E-12
129	SS	RT	4	4	0	0	3.1E-06	TBVG5	9.9E-01	3.1E-06	5.2E-06	1.6E-11
130	SS	RT	4	4	0	0	3.1E-06	TBVG9	1.0E-02	3.1E-08	2.0E-03	6.2E-11
131	SS	RT	4	4	0	0	3.1E-06	TBVG9	3.6E-03	1.1E-08	2.0E-03	2.2E-11
132	SS	RT	4	1	0	0	1.6E-05	TBVG9	9.9E-01	1.6E-05	2.0E-03	3.2E-08
133	SS	RT	4	1	0	0	1.6E-05	TBVG9	1.0E-02	1.6E-07	2.0E-03	3.2E-10
134	SL	RT	5	0	0	0	4.5E-04	TBVG5	9.9E-01	4.5E-04	5.2E-06	2.3E-09
135	SL	RT	5	0	0	0	4.5E-04	TBVG5	1.0E-02	4.5E-06	5.2E-06	2.3E-11
136	SL	RT	5	0	0	1	1.1E-05	TBVG10	9.9E-01	1.1E-05	1.0E-05	1.1E-10
137	SL	RT	5	0	0	1	1.1E-05	TBVG10	1.0E-02	1.1E-07	1.0E-05	1.1E-12
138	SL	RT	5	0	0	2	2.8E-05	TBVG10	9.9E-01	2.8E-05	1.0E-05	2.8E-10
139	SL	RT	5	0	0	2	2.8E-05	TBVG10	1.0E-02	2.8E-07	1.0E-05	2.8E-12
140	SS	RT	5	4	0	0	2.2E-06	TBVG5	9.9E-01	2.2E-06	5.2E-06	1.1E-11
141	SS	RT	5	4	0	0	2.2E-06	TBVG9	1.0E-02	2.2E-08	2.0E-03	4.4E-11
142	SS	RT	5	4	0	0	2.2E-06	TBVG9	3.6E-03	7.9E-09	2.0E-03	1.6E-11
143	SS	RT	5	1	0	0	1.4E-05	TBVG9	9.9E-01	1.4E-05	2.0E-03	2.8E-08
144	SS	RT	5	1	0	0	1.4E-05	TBVG9	1.0E-02	1.4E-07	2.0E-03	2.8E-10
145	SL	RT	6	0	0	0	5.0E-04	MSLB1	9.9E-01	5.0E-04	6.2E-04	3.1E-07
146	SL	RT	6	0	0	0	5.0E-04	TBVG9	1.0E-02	5.0E-06	2.0E-03	1.0E-08
147	SL	RT	6	0	0	1	1.2E-05	MSLB7	9.9E-01	1.2E-05	6.2E-04	7.4E-09
148	SL	RT	6	0	0	1	1.2E-05	TBVG9	1.0E-02	1.2E-07	2.0E-03	2.4E-10
149	SL	RT	6	0	0	2	3.1E-05	MSLB7	9.9E-01	3.1E-05	6.2E-04	1.9E-08
150	SL	RT	6	0	0	2	3.1E-05	TBVG9	1.0E-02	3.1E-07	2.0E-03	6.2E-10
151	SS	RT	6	4	0	0	2.5E-06	MSLB1	9.9E-01	2.5E-06	6.2E-04	1.6E-09
152	SS	RT	6	4	0	0	2.5E-06	TBVG9	1.0E-02	2.5E-08	2.0E-03	5.0E-11
153	SS	RT	6	4	0	0	2.5E-06	TBVG9	3.6E-03	9.0E-09	2.0E-03	1.8E-11
154	SS	RT	6	1	0	0	1.6E-05	MSLB7	9.9E-01	1.6E-05	6.2E-04	9.9E-09
155	SS	RT	6	1	0	0	1.6E-05	TBVG9	1.0E-02	1.6E-07	2.0E-03	3.2E-10
156	RES	RT					1.9E-04	LANL10	1.0E+00	1.9E-04	5.4E-03	1.0E-06
157	FW	EMFW	0	1	0	0	8.4E-02	OVRFD3	1.0E+00	8.4E-02	1.0E-07	8.4E-09
158	FW	EMFW	0	3	0	0	1.4E-04	OVRFD6	9.9E-01	1.4E-04	1.4E-06	2.0E-10
159	FW	EMFW	0	3	0	0	1.4E-04	OVRFD6	1.0E-02	1.4E-06	1.4E-06	2.0E-12

REC	TYPE	INIT	SPC	FWS	ECC	PPC	EFREQ	TRANS	HF	TFREQ	P(F|T)	P(TWC)
160	FW	EMFW	0	3	0	1	2.9E-06	PSBG4	9.9E-01	2.9E-06	1.3E-03	3.8E-09
161	FW	EMFW	0	3	0	1	2.9E-06	PSBG4	1.0E-02	2.9E-08	1.3E-03	3.8E-11
162	FW	EMFW	0	3	0	2	6.8E-06	PSBG4	9.9E-01	6.7E-06	1.3E-03	8.7E-09
163	FW	EMFW	0	3	0	2	6.8E-06	PSBG4	1.0E-02	6.8E-08	1.3E-03	8.8E-11
164	FW	EMFW	0	4	0	0	5.5E-04	OVRFD6	9.9E-01	5.5E-04	1.4E-06	7.7E-10
165	FW	EMFW	0	4	0	0	5.5E-04	OVRFD6	1.0E-02	5.5E-08	1.4E-06	7.7E-12
166	FW	EMFW	0	4	0	0	5.5E-04	OVRFD7	3.6E-03	2.0E-06	1.0E-07	2.0E-13
167	FW	EMFW	0	4	0	1	1.3E-05	PSBG3	9.9E-01	1.3E-05	1.0E-07	1.3E-12
168	FW	EMFW	0	4	0	1	1.3E-05	PSBG4	1.0E-02	1.3E-07	1.3E-03	1.7E-10
169	FW	EMFW	0	4	0	1	1.3E-05	PSBG2	3.6E-03	4.7E-08	1.0E-07	4.7E-15
170	FW	EMFW	0	4	0	2	3.3E-05	PSBG3	9.9E-01	3.3E-05	1.0E-07	3.3E-12
171	FW	EMFW	0	4	0	2	3.3E-05	PSBG4	1.0E-02	3.3E-07	1.3E-03	4.3E-10
172	FW	EMFW	0	4	0	2	3.3E-05	PSBG2	3.6E-03	1.2E-07	1.0E-07	1.2E-14
173	FW	EMFW	0	2	0	0	3.1E-04	OVRFD5	9.9E-01	3.1E-04	1.0E-07	3.1E-11
174	FW	EMFW	0	2	0	0	3.1E-04	OVRFD5	1.0E-02	3.1E-06	1.0E-07	3.1E-12
175	FW	EMFW	0	2	0	1	7.7E-06	PSBG2	9.9E-01	7.6E-06	1.0E-07	7.6E-13
176	FW	EMFW	0	2	0	1	7.7E-06	PSBG2	1.0E-02	7.7E-08	1.0E-07	7.7E-15
177	FW	EMFW	0	2	0	2	2.0E-05	PSBG2	9.9E-01	2.0E-05	1.0E-07	2.0E-12
178	FW	EMFW	0	2	0	2	2.0E-05	PSBG2	1.0E-02	2.0E-07	1.0E-07	2.0E-14
179	SS	EMFW	1	1	0	0	1.2E-02	TBVG3	9.9E-01	1.2E-02	1.0E-07	1.2E-09
180	SS	EMFW	1	1	0	0	1.2E-02	TBVG4	1.0E-02	1.2E-04	2.0E-03	2.4E-07
181	SS	EMFW	1	1	0	1	2.9E-04	TBVG6	9.9E-01	2.9E-04	1.7E-06	4.9E-10
182	SS	EMFW	1	1	0	1	2.9E-04	TBVG6	1.0E-02	2.9E-06	1.7E-06	4.9E-12
183	SS	EMFW	1	1	0	2	7.7E-04	TBVG6	9.9E-01	7.6E-04	1.7E-06	1.3E-09
184	SS	EMFW	1	1	0	2	7.7E-04	TBVG6	1.0E-02	7.7E-06	1.7E-06	1.3E-11
185	SS	EMFW	1	1	1	0	1.2E-05	TBVG3	9.9E-01	1.2E-05	1.0E-07	1.2E-12
186	SS	EMFW	1	1	1	0	1.2E-05	TBVG4	1.0E-02	1.2E-07	2.0E-03	2.4E-10
187	SS	EMFW	1	3	0	0	1.6E-05	TBVG2	9.9E-01	1.6E-05	1.0E-07	1.6E-12
188	SS	EMFW	1	3	0	0	1.6E-05	LANL7	1.0E-02	1.6E-07	1.8E-03	2.9E-10
189	SS	EMFW	1	4	0	0	8.0E-05	TBVG2	9.9E-01	7.9E-05	1.0E-07	7.9E-12
190	SS	EMFW	1	4	0	0	8.0E-05	TBVG1	1.0E-02	8.0E-07	4.0E-07	3.2E-13
191	SS	EMFW	1	4	0	0	8.0E-05	TBVG1	3.6E-03	2.9E-07	4.0E-07	1.2E-13
192	SS	EMFW	1	4	0	2	3.5E-06	TBVG10	9.9E-01	3.5E-06	1.0E-05	3.5E-11
193	SS	EMFW	1	4	0	2	3.5E-06	TBVG10	1.0E-02	3.5E-08	1.0E-05	3.5E-13
194	SS	EMFW	1	4	0	2	3.5E-06	TBVG4	3.6E-03	1.3E-08	2.0E-03	2.6E-11
195	SS	EMFW	1	2	0	0	4.7E-05	TBVG4	9.9E-01	4.7E-05	2.0E-03	9.4E-08
196	SS	EMFW	1	2	0	0	4.7E-05	TBVG4	1.0E-02	4.7E-07	2.0E-03	9.4E-10
197	SS	EMFW	1	2	0	2	2.7E-06	TBVG6	9.9E-01	2.7E-06	1.7E-06	4.6E-12
198	SS	EMFW	1	2	0	2	2.7E-06	TBVG6	1.0E-02	2.7E-08	1.7E-06	4.6E-14
199	SS	EMFW	2	1	0	0	1.4E-04	TBVG3	9.9E-01	1.4E-04	1.0E-07	1.4E-11
200	SS	EMFW	2	1	0	0	1.4E-04	TBVG4	1.0E-02	1.4E-06	2.0E-03	2.8E-09
201	SS	EMFW	2	1	0	1	3.1E-06	TBVG6	9.9E-01	3.1E-06	1.7E-06	5.3E-12
202	SS	EMFW	2	1	0	1	3.1E-06	TBVG6	1.0E-02	3.1E-08	1.7E-06	5.3E-14
203	SS	EMFW	2	1	0	2	7.2E-06	TBVG6	9.9E-01	7.1E-06	1.7E-06	1.2E-11
204	SS	EMFW	2	1	0	2	7.2E-06	TBVG6	1.0E-02	7.2E-08	1.7E-06	1.2E-13
205	SS	EMFW	3	1	0	0	1.3E-03	TBVG5	9.9E-01	1.3E-03	5.2E-06	6.7E-09
206	SS	EFFW	3	1	0	0	1.3E-03	TBVG9	1.0E-02	1.3E-05	2.0E-03	2.6E-08
207	SS	EMFW	3	1	0	1	3.2E-05	TBVG10	9.9E-01	3.2E-05	1.0E-05	3.2E-10
208	SS	EMFW	3	1	0	1	3.2E-05	TBVG10	1.0E-02	3.2E-07	1.0E-05	3.2E-12
209	SS	EMFW	3	1	0	2	8.4E-05	TBVG10	9.9E-01	8.3E-05	1.0E-05	8.3E-10
210	SS	EMFW	3	1	0	2	8.4E-05	TBVG10	1.0E-02	8.4E-07	1.0E-05	8.4E-12
211	SS	EMFW	3	1	1	0	1.4E-06	TBVG5	9.9E-01	1.4E-06	5.2E-06	7.3E-12
212	SS	EMFW	3	1	1	0	1.4E-06	TBVG9	1.0E-02	1.4E-08	2.0E-03	2.8E-11

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REC TYPE INIT   SPC FWS ECC PPC EFREO    TRANS   HF       TFREQ    P(F|T)   P(TWC)
213 SS   EMFW   3   3   0   0   1.8E-06  TBVG5   9.9E-01  1.8E-06  5.2E-06  9.4E-12
214 SS   EMFW   3   3   0   0   1.8E-06  LANL7   1.0E-02  1.8E-08  1.8E-03  3.2E-11
215 SS   EMFW   3   4   0   0   7.5E-06  TBVC5   9.9E-01  7.4E-06  5.2E-06  3.8E-11
216 SS   EMFW   3   4   0   0   7.5E-06  TBVG5   1.0E-02  7.5E-08  5.2E-06  3.9E-13
217 SS   EMFW   3   4   0   0   7.5E-06  TBVG9   3.6E-03  2.7E-08  2.0E-03  5.4E-11
218 SS   EMFW   3   2   0   0   4.7E-06  TBVG9   9.9E-01  4.7E-06  2.0E-03  9.4E-09
219 SS   EMFW   3   2   0   0   4.7E-06  TBVG9   1.0E-02  4.7E-08  2.0E-03  9.4E-11
220 SS   EMFW   4   1   0   0   8.6E-06  TBVG5   9.9E-01  8.5E-06  5.2E-06  4.4E-11
221 SS   EMFW   4   1   0   0   8.6E-06  TBVG9   1.0E-02  8.6E-08  2.0E-03  1.7E-10
222 SS   EMFW   5   1   0   0   6.3E-06  TBVG5   9.9E-01  6.2E-06  5.2E-06  3.2E-11
223 SS   EMFW   5   1   0   0   6.3E-06  TBVG9   1.0E-02  6.3E-08  2.0E-03  1.3E-10
224 SS   EMFW   6   1   0   0   7.0E-06  MSLB7   9.9E-01  6.9E-06  6.2E-04  4.3E-09
225 SS   EMFW   6   1   0   0   7.0E-06  TBVG9   1.0E-02  7.0E-08  2.0E-03  1.4E-10
226 RES  EMFW                   7.2E-05  LANL7   1.0E+00  7.2E-05  1.8E-03  1.3E-07
227 SL   LSLB   1   0   0   0   9.6E-04  MSLB1   9.9E-01  9.5E-04  6.2E-04  5.9E-07
228 SL   LSLB   1   0   0   0   9.6E-04  MSLB4   1.0E-02  9.6E-06  3.0E-03  2.9E-08
229 SL   LSLB   1   1   0   0   3.0E-05  MSLB3   9.9E-01  3.0E-05  4.0E-04  1.2E-08
230 SL   LSLB   1   1   0   0   3.0E-05  TBVG9   1.0E-02  3.0E-07  2.0E-03  6.0E-10
231 SL   LSLB   3   0   0   0   5.5E-06  MSLB1   9.9E-01  5.4E-06  6.2E-04  3.3E-09
232 SL   LSLB   3   0   0   0   5.5E-06  TBVG9   1.0E-02  5.5E-08  2.0E-03  1.1E-10
233 RES  LSLB                   5.3E-06  MSLB4   1.0E+00  5.3E-06  3.0E-03  1.6E-08
234 SL   SSLB   1   0   0   0   9.0E-03  TBVG2   9.9E-01  8.9E-03  1.0E-07  8.9E-10
235 SL   SSLB   1   0   0   0   9.0E-03  TBVG1   1.0E-02  9.0E-05  4.0E-07  3.6E-11
236 SL   SSLB   1   1   0   0   2.8E-04  TBVG3   9.9E-01  2.8E-04  1.0E-07  2.8E-11
237 SL   SSLB   1   1   0   0   2.8E-04  TBVG4   1.0E-02  2.8E-06  2.0E-03  5.6E-09
238 SS   SSLB   1   4   0   0   2.4E-06  TBVG2   9.9E-01  2.4E-06  1.0E-07  2.4E-13
239 SS   SSLB   1   4   0   0   2.4E-06  TBVG1   1.0E-02  2.4E-08  1.4E-06  3.4E-14
240 SS   SSLB   1   4   0   0   2.4E-06  TBVG1   3.6E-03  8.6E-09  4.0E-07  3.5E-15
241 SL   SSLB   3   0   0   0   6.3E-04  TBVG5   9.9E-01  6.2E-04  5.2E-06  3.2E-09
242 SL   SSLB   3   0   0   0   6.3E-04  TBVG5   1.0E-02  6.3E-06  5.2E-06  3.3E-11
243 SL   SSLB   3   0   0   1   1.5E-05  TBVG10  9.9E-01  1.5E-05  1.0E-05  1.5E-10
244 SL   SSLB   3   0   0   1   1.5E-05  TBVG10  1.0E-02  1.5E-07  1.0E-05  1.5E-12
245 SL   SSLB   3   0   0   2   4.0E-05  TBVG10  9.9E-01  4.0E-05  1.0E-05  4.0E-10
246 SL   SSLB   3   0   0   2   4.0E-05  TBVG10  1.0E-02  4.0E-07  1.0E-05  4.0E-12
247 SL   SSLB   3   1   0   0   2.0E-05  TBVG9   9.9E-01  2.0E-05  2.0E-03  4.0E-08
248 SL   SSLB   3   1   0   0   2.0E-05  TBVG9   1.0E-02  2.0E-07  2.0E-03  4.0E-10
249 SL   SSLB   3   1   0   2   1.0E-06  TBVG10  9.9E-01  1.0E-06  1.0E-05  1.0E-11
250 SL   SSLB   3   1   0   2   1.0E-06  TBVG10  1.0E-02  1.0E-08  1.0E-05  1.0E-13
251 SL   SSLB   4   0   0   0   6.5E-06  TBVG2   9.9E-01  6.4E-06  1.0E-07  6.4E-13
252 SL   SSLB   4   0   0   0   6.5E-06  TBVG1   1.0E-02  6.5E-08  4.0E-07  2.6E-14
253 RES  SSLB                   1.4E-05  TBVG9   1.0E+00  1.4E-05  2.0E-03  2.8E-08
254 NORM LOMFW  0   0   0   0   5.5E-01  NONE    1.0E+00  5.5E-01  0.0E+00  0.0E+00
255 LOCA LOMFW  0   0   0   1   1.4E-02  PSBG3   1.0E+00  1.4E-02  1.0E-07  1.4E-09
256 LOCA LOMFW  0   0   0   2   3.6E-02  PSBG3   1.0E+00  3.6E-02  1.0E-07  3.6E-09
257 FW   LOMFW  0   3   0   0   2.0E-03  INEL3   9.9E-01  2.0E-03  1.0E-07  2.0E-10
258 FW   LOMFW  0   3   0   0   2.0E-03  INEL3   1.0E-02  2.0E-05  1.0E-07  2.0E-12
259 FW   LOMFW  0   3   0   1   5.1E-05  PSBG5   9.9E-01  5.1E-05  1.0E-07  5.1E-12
260 FW   LOMFW  0   3   0   1   5.1E-05  PSBG4   1.0E-02  5.1E-07  1.3E-03  6.6E-10
261 FW   LOMFW  0   3   0   2   1.3E-04  PSBG5   9.9E-01  1.3E-04  1.0E-07  1.3E-11
262 FW   LOMFW  0   3   0   2   1.3E-04  PSBG4   1.0E-02  1.3E-06  1.3E-03  1.7E-09
263 FW   LOMFW  0   4   0   0   3.9E-03  NONE    9.9E-01  3.9E-03  0.0E+00  0.0E+00
264 FW   LOMFW  0   4   0   0   3.9E-03  OVRFD6  1.0E-02  3.9E-05  1.4E-06  5.5E-11
265 FW   LOMFW  0   4   0   0   3.9E-03  OVRFD7  3.6E-03  1.4E-05  1.0E-07  1.4E-12
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REC	TYPE	INIT	SPC	FWS	ECC	PPC	EFREQ	TRANS	HF	TFREQ	P(F|T)	P(TWC)
266	FW	LOMFW	0	4	0	1	9.5E-05	PSBG3	9.9E-01	9.4E-05	1.0E-07	9.4E-12
267	FW	LOMFW	0	4	0	1	9.5E-05	PSBG4	1.0E-02	9.5E-07	1.3E-03	1.2E-09
268	FW	LOMFW	0	4	0	1	9.5E-05	PSBG2	3.6E-03	3.4E-07	1.0E-07	3.4E-14
269	FW	LOMFW	0	4	0	2	2.5E-04	PSBG3	9.9E-01	2.5E-04	1.0E-07	2.5E-11
270	FW	LOMFW	0	4	0	2	2.5E-04	PSBG4	1.0E-02	2.5E-06	1.3E-03	3.3E-09
271	FW	LOMFW	0	4	0	2	2.5E-04	PSBG2	3.6E-03	9.0E-07	1.0E-07	9.0E-14
272	SL	LOMFW	1	0	0	0	8.4E-02	TBVG2	9.9E-01	8.3E-02	1.0E-07	8.3E-09
273	SL	LOMFW	1	0	0	0	8.4E-02	TBVG1	1.0E-02	8.4E-04	4.0E-07	3.4E-10
274	SL	LOMFW	1	0	0	1	2.1E-03	TBVG6	9.9E-01	2.1E-03	1.7E-06	3.6E-09
275	SL	LOMFW	1	0	0	1	2.1E-03	TBVG6	1.0E-02	2.1E-05	1.7E-06	3.6E-11
276	SL	LOMFW	1	0	0	2	5.5E-03	TBVG6	9.9E-01	5.5E-03	1.7E-06	9.4E-09
277	SL	LOMFW	1	0	0	2	5.5E-03	TBVG6	1.0E-02	5.5E-05	1.7E-06	9.4E-11
278	SL	LOMFW	1	0	1	0	1.1E-04	TBVG2	9.9E-01	1.1E-04	1.0E-07	1.1E-11
279	SL	LOMFW	1	0	1	0	1.1E-04	TBVG1	1.0E-02	1.1E-06	4.0E-07	4.4E-13
280	SL	LOMFW	1	0	1	1	2.7E-06	TBVG6	9.9E-01	2.7E-06	1.7E-06	4.6E-12
281	SL	LOMFW	1	0	1	1	2.7E-06	TBVG6	1.0E-02	2.7E-08	1.7E-06	4.6E-14
282	SL	LOMFW	1	0	1	2	6.2E-06	TBVG6	9.9E-01	6.1E-06	1.7E-06	1.0E-11
283	SL	LOMFW	1	0	1	2	6.2E-06	TBVG6	1.0E-02	6.2E-08	1.7E-06	1.1E-13
284	SS	LOMFW	1	3	0	0	3.1E-04	TBVG2	9.9E-01	3.1E-04	1.0E-07	3.1E-11
285	SS	LOMFW	1	3	0	0	3.1E-04	LANL7	1.0E-02	3.1E-06	1.8E-03	5.6E-09
286	SS	LOMFW	1	3	0	1	7.0E-06	PSBG5	9.9E-01	6.9E-06	1.0E-07	6.9E-13
287	SS	LOMFW	1	3	0	1	7.0E-06	TBVG6	1.0E-02	7.0E-08	1.7E-06	1.2E-13
288	SS	LOMFW	1	3	0	2	1.6E-05	PSBG5	9.9E-01	1.6E-05	1.0E-07	1.6E-12
289	SS	LOMFW	1	3	0	2	1.6E-05	TBVG6	1.0E-02	1.6E-07	1.7E-06	2.7E-13
290	SS	LOMFW	1	4	0	0	5.9E-04	TBVG2	9.9E-01	5.8E-04	1.0E-07	5.8E-11
291	SS	LOMFW	1	4	0	0	5.9E-04	TBVG1	1.0E-02	5.9E-06	4.0E-07	2.4E-12
292	SS	LOMFW	1	4	0	0	5.9E-04	TBVG1	3.6E-03	2.1E-06	4.0E-07	8.5E-13
293	SS	LOMFW	1	4	0	1	1.5E-05	PSBG5	9.9E-01	1.5E-05	1.0E-07	1.5E-12
294	SS	LOMFW	1	4	0	1	1.5E-05	TBVG6	1.0E-02	1.5E-07	1.7E-06	2.6E-13
295	SS	LOMFW	1	4	0	1	1.5E-05	TBVG6	3.6E-03	5.4E-08	1.7E-06	9.2E-14
296	SS	LOMFW	1	4	0	2	3.7E-05	PSBG5	9.9E-01	3.7E-05	1.0E-07	3.7E-12
297	SS	LOMFW	1	4	0	2	3.7E-05	TBVG6	1.0E-02	3.7E-07	1.7E-06	6.3E-13
298	SS	LOMFW	1	4	0	2	3.7E-05	TBVG6	3.6E-03	1.3E-07	1.7E-06	2.2E-13
299	SL	LOMFW	2	0	0	0	9.8E-04	TBVG2	9.9E-01	9.7E-04	1.0E-07	9.7E-11
300	SL	LOMFW	2	0	0	0	9.8E-04	TBVG1	1.0E-02	9.8E-06	4.0E-07	3.9E-12
301	SL	LOMFW	2	0	0	1	2.4E-05	TBVG6	9.9E-01	2.4E-05	1.7E-06	4.1E-11
302	SL	LOMFW	2	0	0	1	2.4E-05	TBVG6	1.0E-02	2.4E-07	1.7E-06	4.1E-13
303	SL	LOMFW	2	0	0	2	6.3E-05	TBVG6	9.9E-01	6.2E-05	1.7E-06	1.1E-10
304	SL	LOMFW	2	0	0	2	6.3E-05	TBVG6	1.0E-02	6.3E-07	1.7E-06	1.1E-12
305	SS	LOMFW	2	3	0	0	3.0E-06	TBVG2	9.9E-01	3.0E-06	1.0E-07	3.0E-13
306	SS	LOMFW	2	3	0	0	3.0E-06	LANL7	1.0E-02	3.0E-08	1.8E-03	5.4E-11
307	SS	LOMFW	2	4	0	0	5.0E-06	TBVG2	9.9E-01	5.0E-06	1.0E-07	5.0E-13
308	SS	LOMFW	2	4	0	0	5.0E-06	TBVG1	1.0E-02	5.0E-08	4.0E-07	2.0E-14
309	SS	LOMFW	2	4	0	0	5.0E-06	TBVG1	3.6E-03	1.8E-08	4.0E-07	7.2E-15
310	SL	LOMFW	3	0	0	0	9.2E-03	TBVG5	9.9E-01	9.1E-03	5.2E-06	4.7E-08
311	SL	LOMFW	3	0	0	0	9.2E-03	TBVG5	1.0E-02	9.2E-05	5.2E-06	4.8E-08
312	SL	LOMFW	3	0	0	1	2.3E-04	TBVG10	9.9E-01	2.3E-04	1.0E-05	2.3E-09
313	SL	LOMFW	3	0	0	1	2.3E-04	TBVG10	1.0E-02	2.3E-06	1.0E-05	2.3E-11
314	SL	LOMFW	3	0	0	2	6.0E-04	TBVG10	9.9E-01	5.9E-04	1.0E-05	5.9E-09
315	SL	LOMFW	3	0	0	2	6.0E-04	TBVG10	1.0E-02	6.0E-06	1.0E-05	6.0E-11
316	SL	LOMFW	3	0	1	0	1.2E-05	TBVG5	9.9E-01	1.2E-05	5.2E-06	6.2E-11
317	SL	LOMFW	3	0	1	0	1.2E-05	TBVG5	1.0E-02	1.2E-07	5.2E-06	6.2E-13
318	SS	LOMFW	3	3	0	0	3.4E-05	TBVG5	9.9E-01	3.4E-05	5.2E-06	1.8E-10

REC	TYPE	INIT	SPC	FWS	ECC	PPC	EFREQ	TRANS	HF	TFREQ	P(F|T)	P(TWC)
319	SS	LOMFW	3	3	0	0	3.4E-05	LANL7	1.0E-02	3.4R-07	1.8E-03	6.1E-10
320	SS	LOMFW	3	3	0	2	1.8E-06	TBVG10	9.9E-01	1.8E-06	1.0E-05	1.8E-11
321	SS	LOMFW	3	3	0	2	1.8E-06	TBVG10	1.0E-02	1.8E-08	1.0E-05	1.8E-13
322	SS	LOMFW	3	4	0	0	6.5E-05	TBVG5	9.9E-01	6.4E-05	5.2E-06	3.3E-10
323	SS	LOMFW	3	4	0	0	6.5E-05	TBVG9	1.0E-02	6.5E-07	2.0E-03	1.3E-09
324	SS	LOMFW	3	4	0	0	6.5E-05	TBVG9	3.6E-03	2.3E-07	2.0E-03	4.6E-10
325	SS	LOMFW	3	4	0	1	1.3E-06	TBVG10	9.9E-01	1.3E-06	1.0E-05	1.3E-11
326	SS	LOMFW	3	4	0	1	1.3E-06	TBVG10	1.0E-02	1.3E-08	1.0E-05	1.3E-13
327	SS	LOMFW	3	4	0	1	1.3E-06	TBVG10	3.6E-03	4.7E-09	1.0E-05	4.7E-14
328	SS	LOMFW	3	4	0	2	3.0E-06	TBVG10	9.9E-01	3.0E-06	1.0E-05	3.0E-11
329	SS	LOMFW	3	4	0	2	3.0E-06	TBVG10	1.0E-02	3.0E-08	1.0E-05	3.0E-13
330	SS	LOMFW	3	4	0	2	3.0E-06	TBVG10	3.6E-03	1.1E-08	1.0E-05	1.1E-13
331	SL	LOMFW	4	0	0	0	8.3E-05	TBVG5	9.9E-01	8.2E-05	5.2E-06	4.3E-10
332	SL	LOMFW	4	0	0	0	8.3E-05	TBVG5	1.0E-02	8.3E-07	5.2E-06	4.3E-12
333	SL	LOMFW	4	0	0	2	4.3E-06	TBVG10	9.9E-01	4.3E-06	1.0E-05	4.3E-11
334	SL	LOMFW	4	0	0	2	4.3E-06	TBVG10	1.0E-02	4.3E-08	1.0E-05	4.3E-13
335	SL	LOMFW	5	0	0	0	5.4E-05	TBVG5	9.9E-01	5.3E-05	5.2E-06	2.8E-10
336	SL	LOMFW	5	0	0	0	5.4E-05	TBVG6	1.0E-02	5.4E-07	5.2E-06	2.8E-12
337	SL	LOMFW	5	0	0	1	1.3E-06	TBVG10	9.9E-01	1.3E-06	1.0E-05	1.3E-11
338	SL	LOMFW	5	0	0	1	1.3E-06	TBVG10	1.0E-02	1.3E-08	1.0E-05	1.3E-13
339	SL	LOMFW	5	0	0	2	3.1E-06	TBVG10	9.9E-01	3.1E-06	1.0E-05	3.1E-11
340	SL	LOMFW	5	0	0	2	3.1E-06	TBVG10	1.0E-02	3.1E-08	1.0E-05	3.1E-13
341	SL	LOMFW	6	0	0	0	6.1E-05	MSLB1	9.9E-01	6.0E-05	6.2E-04	3.7E-08
342	SL	LOMFW	6	0	0	0	6.1E-05	MSLB1	9.9E-01	6.0E-05	6.2E-04	3.7E-08
343	SL	LOMFW	6	0	0	0	6.1E-05	TBVG9	1.0E-02	6.1E-07	2.0E-03	1.2E-09
344	SL	LOMFW	6	0	0	1	1.5E-06	MSLB7	9.9E-01	1.5E-06	6.2E-04	9.3E-10
345	SL	LOMFW	6	0	0	1	1.5E-06	TBVG9	1.0E-02	1.5E-08	2.0E-03	3.0E-11
346	SL	LOMFW	6	0	0	2	3.5E-06	MSLB7	9.9E-01	3.5E-06	6.2E-04	2.2E-09
347	SL	LOMFW	6	0	0	2	3.5E-06	TBVG9	1.0E-02	3.5E-08	2.0E-03	7.0E-11
348	RES	LOMFW					5.9E-05	TBVG9	1.0E+00	5.9E-05	2.0E-03	1.2E-07
349	LOCA	SBL1	0	0	0	0	7.6E-02	PSBG1	1.0E+00	7.6E-02	1.0E-07	7.6E-09
350	LOCA	SBL1	0	0	0	1	1.9E-03	PSBG1	1.0E+00	1.9E-03	1.0E-07	1.9E-10
351	LOCA	SBL1	0	0	0	2	5.0E-03	PSBG1	1.0E+00	5.0E-03	1.0E-07	5.0E-10
352	LOCA	SBL1	0	0	1	0	9.9E-05	PSBG1	1.0E+00	9.9E-05	1.0E-07	9.9E-12
353	LOCA	SBL1	0	0	1	1	2.4E-06	PSBG1	1.0E+00	2.4E-06	1.0E-07	2.4E-13
354	LOCA	SBL1	0	0	1	2	5.5E-06	PSBG1	1.0E+00	5.5E-06	1.0E-07	5.5E-13
355	LOCA	SBL1	0	1	0	0	2.4E-03	PSBG2	1.0E+00	2.4E-03	1.0E-07	2.4E-10
356	LOCA	SBL1	0	1	0	1	5.9E-05	PSBG2	1.0E+00	5.9E-05	1.0E-07	5.9E-12
357	LOCA	SBL1	0	1	0	2	1.6E-04	PSBG2	1.0E+00	1.6E-04	1.0E-07	1.6E-11
358	LOCA	SBL1	0	1	1	0	2.5E-06	PSBG2	1.0E+00	2.5E-06	1.0E-07	2.5E-13
359	LOCA	SBL1	0	3	0	0	6.6E-06	PSBG2	9.9E-01	6.5E-06	1.0E-07	6.5E-13
360	LOCA	SBL1	0	3	0	0	6.6E-06	PSBG4	1.0E-02	6.6E-08	1.3E-03	8.6E-11
361	LOCA	SBL1	0	4	0	0	3.1E-05	NONE	9.9E-01	3.1E-05	0.0E+00	0.0E+00
362	LOCA	SBL1	0	4	0	0	3.1E-05	OVRFD6	1.0E-02	3.1E-07	1.4E-06	4.3E-13
363	LOCA	SBL1	0	4	0	0	3.1E-05	OVRFD7	3.6E-03	1.1E-07	1.0E-07	1.1E-14
364	LOCA	SBL1	0	2	0	0	8.7E-06	PSBG2	9.9E-01	8.6E-06	1.0E-07	8.6E-13
365	LOCA	SBL1	0	2	0	0	8.7E-06	PSBG2	1.0E-02	8.7E-08	1.0E-07	8.7E-15
366	LOCA	SBL1	1	0	0	0	1.2E-02	PSBG5	9.9E-01	1.2E-02	1.0E-07	1.2E-09
367	LOCA	SBL1	1	0	0	0	1.2E-02	TBVG6	1.0E-02	1.2E-04	1.7E-06	2.0E-10
368	LOCA	SBL1	1	0	0	1	2.9E-04	PSBG5	9.9E-01	2.9E-04	1.0E-07	2.9E-11
369	LOCA	SBL1	1	0	0	1	2.9E-04	TBVG6	1.0E-02	2.9E-06	1.7E-06	4.9E-12
370	LOCA	SBL1	1	0	0	2	7.5E-04	PSBG5	9.9E-01	7.4E-04	1.0E-07	7.4E-11
371	LOCA	SBL1	1	0	0	2	7.5E-04	TBVG6	1.0E-02	7.5E-06	1.7E-06	1.3E-11

REC	TYPE	INIT	SPC	FWS	ECC	PPC	EFREQ	TRANS	HF	TFREQ	P(F|T)	P(TWC)
372	LOCA	SBL1	1	0	1	0	1.3E-05	PSBG5	9.9E-01	1.3E-05	1.0E-07	1.3E-12
373	LOCA	SBL1	1	0	1	0	1.3E-05	TBVG6	1.0E-02	1.3E-07	1.7E-06	2.2E-13
374	LOCA	SBL1	1	3	0	0	1.7E-05	PSBG5	9.9E-01	1.7E-05	1.0E-07	1.7E-12
375	LOCA	SBL1	1	3	0	0	1.7E-05	TBVG6	1.0E-02	1.7E-07	1.7E-06	2.9E-13
376	LOCA	SBL1	1	4	0	0	7.7E-05	PSBG5	9.9E-01	7.6E-05	1.0E-07	7.6E-12
377	LOCA	SBL1	1	4	0	0	7.7E-05	TBVG6	1.0E-02	7.7E-07	1.7E-06	1.3E-12
378	LOCA	SBL1	1	4	0	2	3.6E-06	PSBG5	9.9E-01	3.6E-06	1.0E-07	3.6E-13
379	LOCA	SBL1	1	4	0	2	3.6E-06	TBVG6	1.0E-02	3.6E-08	1.7E-06	6.1E-14
380	LOCA	SBL1	1	4	0	2	3.6E-06	TBVG6	3.6E-03	1.3E-08	1.7E-06	2.2E-14
381	LOCA	SBL1	1	1	0	0	3.7E-04	TBVG6	9.9E-01	3.7E-04	1.7E-06	6.3E-10
382	LOCA	SBL1	1	1	0	0	3.7E-04	TBVG6	1.0E-02	3.7E-06	1.7E-06	6.3E-12
383	LOCA	SBL1	1	1	0	1	8.2E-06	TBVG6	9.9E-01	8.1E-06	1.7E-06	1.4E-11
384	LOCA	SBL1	1	1	0	1	8.2E-06	TBVG6	1.0E-02	8.2E-08	1.7E-06	1.4E-13
385	LOCA	SBL1	1	1	0	2	2.1E-05	TBVG6	9.9E-01	2.1E-05	1.7E-06	3.6E-11
386	LOCA	SBL1	1	1	0	2	2.1E-05	TBVG6	1.0E-02	2.1E-07	1.7E-06	3.6E-13
387	LOCA	SBL1	2	0	0	0	1.4E-04	PSBG5	9.9E-01	1.4E-04	1.0E-07	1.4E-11
388	LOCA	SBL1	2	0	0	0	1.4E-04	TBVG6	1.0E-02	1.4E-06	1.7E-06	2.4E-12
389	LOCA	SBL1	2	0	0	2	7.6E-06	PSBG5	9.9E-01	7.5E-06	1.0E-07	7.5E-13
390	LOCA	SBL1	2	0	0	2	7.6E-06	TBVG6	1.0E-02	7.6E-08	1.7E-06	1.3E-13
391	LOCA	SBL1	2	1	0	0	3.5E-06	TBVG6	9.9E-01	3.5E-06	1.7E-06	6.0E-12
392	LOCA	SBL1	2	1	0	0	3.5E-06	TBVG6	1.0E-02	3.5E-08	1.7E-06	6.0E-14
393	LOCA	SBL1	3	0	0	0	1.3E-03	TBVG10	9.9E-01	1.3E-03	1.0E-05	1.3E-08
394	LOCA	SBL1	3	0	0	0	1.3E-03	TBVG10	1.0E-02	1.3E-05	1.0E-05	1.3E-10
395	LOCA	SBL1	3	0	0	1	3.0E-05	TBVG10	9.9E-01	3.0E-05	1.0E-05	3.0E-10
396	LOCA	SBL1	3	0	0	1	3.0E-05	TBVG10	1.0E-02	3.0E-07	1.0E-05	3.0E-12
397	LOCA	SBL1	3	0	0	2	8.3E-05	TBVG10	9.9E-01	8.2E-05	1.0E-05	8.2E-10
398	LOCA	SBL1	3	0	0	2	8.3E-05	TBVG10	1.0E-02	8.3E-07	1.0E-05	8.3E-12
399	LOCA	SBL1	3	0	1	0	1.4E-06	TBVG10	9.9E-01	1.4E-06	1.0E-05	1.4E-11
400	LOCA	SBL1	3	0	1	0	1.4E-06	TBVG10	1.0E-02	1.4E-08	1.0E-05	1.4E-13
401	LOCA	SBL1	3	3	0	0	1.9E-06	TBVG10	9.9E-01	1.9E-06	1.0E-05	1.9E-11
402	LOCA	SBL1	3	3	0	0	1.9E-06	TBVG10	1.0E-02	1.9E-08	1.0E-05	1.9E-13
403	LOCA	SBL1	3	4	0	0	7.8E-06	TBVG10	9.9E-01	7.7E-06	1.0E-05	7.7E-11
404	LOCA	SBL1	3	4	0	0	7.8E-06	TBVG10	1.0E-02	7.8E-08	1.0E-05	7.8E-13
405	LOCA	SBL1	3	4	0	0	7.8E-06	TBVG10	3.6E-03	2.8E-08	1.0E-05	2.8E-13
406	LOCA	SBL1	3	1	0	0	4.0E-05	TBVG10	9.9E-01	4.0E-05	1.0E-05	4.0E-10
407	LOCA	SBL1	3	1	0	0	4.0E-05	TBVG10	1.0E-02	4.0E-07	1.0E-05	4.0E-12
408	LOCA	SBL1	3	1	0	2	2.1E-06	TBVG10	9.9E-01	2.1E-06	1.0E-05	2.1E-11
409	LOCA	SBL1	3	1	0	2	2.1E-06	TBVG10	1.0E-02	2.1E-08	1.0E-05	2.1E-13
410	LOCA	SBL1	4	0	0	0	9.1E-06	TBVG10	9.9E-01	9.0E-06	1.0E-05	9.0E-11
411	LOCA	SBL1	4	0	0	0	9.1E-06	TBVG10	1.0E-02	9.1E-08	1.0E-05	9.1E-13
412	LOCA	SBL1	5	0	0	0	6.6E-06	TBVG10	9.9E-01	6.5E-06	1.0E-05	6.5E-11
413	LOCA	SBL1	5	0	0	0	6.6E-06	TBVG10	1.0E-02	6.6E-08	1.0E-05	6.6E-13
414	LOCA	SBL1	6	0	0	0	7.3E-06	MSLB7	9.9E-01	7.2E-06	6.2E-04	4.5E-09
415	LOCA	SBL1	6	0	0	0	7.3E-06	TBVG9	1.0E-02	7.3E-08	2.0E-03	1.5E-10
416	RES	SBL1					7.8E-05	TBVG9	1.0E+00	7.8E-05	2.0E-03	1.6E-07
417	LOCA	SBL2	0	0	0	0	8.2E-03	PSBG1	1.0E+00	8.2E-03	1.0E-07	8.2E-10
418	LOCA	SBL2	0	0	1	0	1.0E-05	PSBG1	1.0E+00	1.0E-05	1.0E-07	1.0E-12
419	LOCA	SBL2	0	1	0	0	2.6E-04	PSBG2	9.9E-01	2.6E-04	1.0E-07	2.6E-11
420	LOCA	SBL2	0	1	0	0	2.6E-04	PSBG4	1.0E-02	2.6E-06	1.3E-03	3.4E-09
421	LOCA	SBL2	0	4	0	0	2.7E-06	NONE	9.9E-01	2.7E-06	0.0E+00	0.0E+00
422	LOCA	SBL2	0	4	0	0	2.7E-06	OVRFD6	1.0E-02	2.7E-08	1.4E-06	3.8E-14
423	LOCA	SBL2	0	4	0	0	2.7E-06	OVRFD7	3.6E-03	9.7E-09	1.0E-07	9.7E-16
424	LOCA	SBL2	0	2	0	0	1.0E-06	PSBG2	1.0E+00	1.0E-06	1.0E-07	1.0E-13

REC	TYPE	INIT	SPC	FWS	ECC	PPC	EFREQ	TRANS	HF	TFREQ	P(F|T)	P(TWC)
425	LOCA	SBL2	1	0	0	0	1.2E-03	PSBG5	9.9E-01	1.2E-03	1.0E-07	1.2E-10
426	LOCA	SBL2	1	0	0	0	1.2E-03	TBVG6	1.0E-02	1.2E-05	1.7E-06	2.0E-11
427	LOCA	SBL2	1	3	0	0	2.1E-06	PSBG5	9.9E-01	2.1E-06	1.0E-07	2.1E-13
428	LOCA	SBL2	1	3	0	0	2.1E-06	TBVG6	1.0E-02	2.1E-08	1.7E-06	3.6E-14
429	LOCA	SBL2	1	4	0	0	6.9E-06	PSBG5	9.9E-01	6.8E-06	1.0E-07	6.8E-13
430	LOCA	SBL2	1	4	0	0	6.9E-06	TBVG6	1.0E-02	6.9E-08	1.7E-06	1.2E-13
431	LOCA	SBL2	1	4	0	0	6.9E-06	TBVG6	3.6E-03	2.5E-08	1.7E-06	4.2E-14
432	LOCA	SBL2	1	1	0	0	4.0E-05	TBVG6	9.9E-01	4.0E-05	1.7E-06	6.8E-11
433	LOCA	SBL2	1	1	0	0	4.0E-05	TBVG6	1.0E-02	4.0E-07	1.7E-06	6.8E-13
434	LOCA	SBL2	2	0	0	0	1.4E-05	PSBG5	9.9E-01	1.4E-05	1.0E-07	1.4E-12
435	LOCA	SBL2	2	0	0	0	1.4E-05	TBVG6	1.0E-02	1.4E-07	1.7E-06	2.4E-13
436	LOCA	SBL2	3	0	0	0	1.4E-04	TBVG10	9.9E-01	1.4E-04	1.0E-05	1.4E-09
437	LOCA	SBL2	3	0	0	0	1.4E-04	TBVG10	1.0E-02	1.4E-06	1.0E-05	1.4E-11
438	LOCA	SBL2	3	1	0	0	4.0E-06	TBVG10	9.9E-01	4.0E-06	1.0E-05	4.0E-11
439	LOCA	SBL2	3	1	0	0	4.0E-06	TBVG10	1.0E-02	4.0E-08	1.0E-05	4.0E-13
440	RES	SBL2					1.8E-05	PSBG4	1.0E+00	1.8E-05	1.3E-03	2.3E-08
441	SGTR	SGTR	0	0	0	0	8.2E-03	SGTR	1.0E+00	8.2E-03	1.0E-07	8.2E-10
442	SGTR	SGTR	0	0	1	0	1.0E-05	SGTR	1.0E+00	1.0E-05	1.0E-07	1.0E-12
443	SGTR	SGTR	0	1	0	0	2.6E-04	PSBG2	9.9E-01	2.6E-04	1.0E-07	2.6E-11
444	SGTR	SGTR	0	1	0	0	2.6E-04	PSBG4	1.0E-02	2.6E-06	1.3E-03	3.4E-09
445	SGTR	SGTR	0	4	0	0	2.7E-06	NONE	9.9E-01	2.7E-06	0.0E+00	0.0E+00
446	SGTR	SGTR	0	4	0	0	2.7E-06	OVRFD6	1.0E-02	2.7E-08	1.4E-06	3.8E-14
447	SGTR	SGTR	0	4	0	0	2.7E-06	OVRFD7	3.6E-03	9.7E-09	1.0E-07	9.7E-16
448	SGTR	SGTR	0	2	0	0	1.0E-06	PSBG2	9.9E-01	1.0E-06	1.0E-07	1.0E-13
449	SGTR	SGTR	0	2	0	0	1.0E-06	PSBG2	1.0E-02	1.0E-08	1.0E-07	1.0E-15
450	SGTR	SGTR	1	0	0	0	1.2E-03	PSBG5	9.9E-01	1.2E-03	1.0E-07	1.2E-10
451	SGTR	SGTR	1	0	0	0	1.2E-03	TBVG6	1.0E-02	1.2E-05	1.7E-06	2.0E-11
452	SGTR	SGTR	1	3	0	0	2.1E-06	PSBG5	9.9E-01	2.1E-06	1.0E-07	2.1E-13
453	SGTR	SGTR	1	3	0	0	2.1E-06	TBVG6	1.0E-02	2.1E-08	1.7E-06	3.6E-14
454	SGTR	SGTR	1	4	0	0	6.9E-06	PSBG5	9.9E-01	6.8E-06	1.0E-07	6.8E-13
455	SGTR	SGTR	1	4	0	0	6.9E-06	TBVG6	1.0E-02	6.9E-08	1.7E-06	1.2E-13
456	SGTR	SGTR	1	4	0	0	6.9E-06	TBVG4	3.6E-03	2.5E-08	2.0E-03	5.0E-11
457	SGTR	SGTR	1	1	0	0	4.0E-05	PSBG5	9.9E-01	4.0E-05	1.0E-07	4.0E-12
458	SGTR	SGTR	1	1	0	0	4.0E-05	TBVG6	1.0E-02	4.0E-07	1.7E-06	6.8E-13
459	SGTR	SGTR	2	0	0	0	1.4E-05	PSBG5	9.9E-01	1.4E-05	1.0E-07	1.4E-12
460	SGTR	SGTR	2	0	0	0	1.4E-05	TBVG6	1.0E-02	1.4E-07	1.7E-06	2.4E-13
461	SGTR	SGTR	3	0	0	0	1.4E-04	TBVG10	9.9E-01	1.4E-04	1.0E-05	1.4E-09
462	SGTR	SGTR	3	0	0	0	1.4E-04	TBVG10	1.0E-02	1.4E-06	1.0E-05	1.4E-11
463	SGTR	SGTR	3	1	0	0	4.0E-06	TBVG10	9.9E-01	4.0E-06	1.0E-05	4.0E-11
464	SGTR	SGTR	3	1	0	0	4.0E-06	TBVG10	1.0E-02	4.0E-08	1.0E-05	4.0E-13
465	RES	SGTR					1.8E-05	TBVG4	1.0E+00	1.8E-05	2.0E-03	3.6E-08
466	NORM	SI	0	0	0	0	7.6E-03	NONE	1.0E+00	7.6E-03	0.0E+00	0.0E+00
467	LOCA	SI	0	0	0	1	1.9E-04	PSBG1	1.0E+00	1.9E-04	1.0E-07	1.9E-11
468	LOCA	SI	0	0	0	2	5.0E-04	PSBG1	1.0E+00	5.0E-04	1.0E-07	5.0E-11
469	FW	SI	0	1	0	0	2.4E-04	OVRFD3	1.0E+00	2.4E-04	1.0E-07	2.4E-11
470	LOCA	SI	0	1	0	1	5.4E-06	PSBG2	9.9E-01	5.3E-06	1.0E-07	5.3E-13
471	LOCA	SI	0	1	0	1	5.4E-06	PSBG2	1.0E-02	5.4E-08	1.0E-07	5.4E-15
472	LOCA	SI	0	1	0	2	1.5E-05	PSBG2	9.9E-01	1.5E-05	1.0E-07	1.5E-12
473	LOCA	SI	0	1	0	2	1.5E-05	PSBG2	1.0E-02	1.5E-07	1.0E-07	1.5E-14
474	FW	SI	0	4	0	0	2.2E-06	NONE	9.9E-01	2.2E-06	0.0E+00	0.0E+00
475	FW	SI	0	4	0	0	2.2E-06	OVRFD6	1.0E-02	2.2E-08	1.4E-06	3.1E-14
476	FW	SI	0	4	0	0	2.2E-06	OVRFD7	3.6E-03	7.9E-09	1.0E-07	7.9E-16
477	SL	SI	1	0	0	0	1.2E-03	TBVG2	9.9E-01	1.2E-03	1.0E-07	1.2E-10

REC	TYPE	INIT	SPC	FWS	ECC	PPC	EFREQ	TRANS	HF	TFREQ	P(F|T)	P(TWC)
478	SL	SI	1	0	0	0	1.2E-03	TBVG1	1.0E-02	1.2E-05	4.0E-07	4.8E-12
479	SL	SI	1	0	0	1	2.8E-05	PSBG5	9.9E-01	2.8E-05	1.0E-07	2.8E-12
480	SL	SI	1	0	0	1	2.8E-05	TBVG6	1.0E-02	2.8E-07	1.7E-06	4.8E-13
481	SL	SI	1	0	0	2	7.2E-05	PSBG5	9.9E-01	7.1E-05	1.0E-07	7.1E-12
482	SL	SI	1	0	0	2	7.2E-05	TBVG6	1.0E-02	7.2E-07	1.7E-06	1.2E-12
483	SS	SI	1	4	0	0	5.7E-06	TBVG2	9.9E-01	5.6E-06	1.0E-07	5.6E-13
484	SS	SI	1	4	0	0	5.7E-06	TBVG1	1.0E-02	5.7E-08	4.0E-07	2.2E-14
485	SS	SI	1	4	0	0	5.7E-06	TBVG1	3.6E-03	2.1E-08	4.0E-07	8.4E-15
486	SS	SI	1	1	0	0	3.6E-05	TBVG3	9.9E-01	3.6E-05	1.0E-07	3.6E-12
487	SS	SI	1	1	0	0	3.6E-05	TBVG4	1.0E-02	3.6E-07	2.0E-03	7.2E-10
488	SL	SI	2	0	0	0	1.2E-05	TBVG2	9.9E-01	1.2E-05	1.0E-07	1.2E-12
489	SL	SI	2	0	0	0	1.2E-05	TBVG1	1.0E-02	1.2E-07	4.0E-07	4.8E-14
490	SL	SI	3	0	0	0	1.3E-04	TBVG5	9.9E-01	1.3E-04	5.2E-06	6.8E-10
491	SL	SI	3	0	0	0	1.3E-04	TBVG5	1.0E-02	1.3E-06	5.2E-06	6.8E-12
492	SL	SI	3	0	0	1	3.1E-06	TBVG10	9.9E-01	3.1E-06	1.0E-05	3.1E-11
493	SL	SI	3	0	0	1	3.1E-06	TBVG10	1.0E-02	3.1E-08	1.0E-05	3.1E-13
494	SL	SI	3	0	0	2	7.1E-06	TBVG10	9.9E-01	7.0E-06	1.0E-05	7.0E-11
495	SL	SI	3	0	0	2	7.1E-06	TBVG10	1.0E-02	7.1E-08	1.0E-05	7.1E-13
496	SS	SI	3	1	0	0	3.3E-06	TBVG9	9.9E-01	3.3E-06	2.0E-03	6.6E-09
497	SS	SI	3	1	0	0	3.3E-06	TBVG9	1.0E-02	3.3E-08	2.0E-03	6.6E-11
498	RES	SI					3.9E-05	TBVG4	1.0E+00	3.9E-05	2.0E-03	7.8E-08

APPENDIX G

RESPONSES TO UTILITY COMMENTS

The material presented in this report is part of an evaluation effort to provide information on the unresolved safety issue of pressurized thermal shock (PTS). Because PTS is an unresolved safety issue, NRC requires that a list of industry comments on the report and the changes which were made as a result of those comments be provided as an appendix to the report. The material presented in this appendix responds to that requirement. The comments are listed and discussed by chapter with comments on the appendices discussed along with the appropriate chapters. Only those comments considered to be pertinent to the PTS study are addressed; that is, comments simply pointing out grammatical or spelling errors are not included.

General Comments

1. This report represents the results of a substantial level of effort over the past three years to evaluate the risk of Pressurized Thermal Shock at Oconee Nuclear Station, Unit 1. As written the report lacks consistency and readability. An enormous wealth of information is provided, much of which is inconsequential to the final result.

Response: We would agree that the report terminology is inconsistent and the readability could be greatly improved. However, it was determined that our efforts would be more wisely spent in completing the Calvert Cliffs and H. B. Robinson reports since these studies reflect a more refined evaluation process. Thus, very little effort was made to improve the readability of the report. We would agree that much of the information presented in the report is inconsequential to the final result. However, it is our opinion that the process used to perform the evaluation and a detailed discussion of the assumptions made in the evaluation is as important if not more so than the actual final result itself. The enormous amount of information is necessary to provide: the approach used to perform the evaluation, the data used in the evaluation, assumptions made in identifying sequences, and our understanding of how the plant responds to certain events. Without this information the quality of the final result can not be determined.

2. The characterization of the frequency (4.5E-6/RY) of a TWC as a best estimate is not appropriate from two perspectives: (1) A review of the report indicates conservatisms that on balance would bias the cited TWC frequency value towards an upper bound and (2) the large uncertainty of a factor of 100 and therefore a range of ten thousand dilutes the confidence of 4.5E-6/RY as being a best estimate.

Response: After the completion of the Calvert Cliffs and H. B. Robinson analysis, the Oconee evaluation was reviewed in light of the improvements in the evaluation techniques. Both conservative and non-conservative factors were identified in the Oconee analysis. As stated in the comment, it did appear that the balance of these factors appeared to be in the conservative direction. However, without redoing the calculations the actual best estimate value could not be determined. The factor of 100 given for the uncertainty value is a 95% value. This value may be somewhat high due to the unrealistic uncertainties used in the negative direction of the temperature profiles and the double counting effect for very high flaw density values. However, in light of other factors, use of error factors as 99% values rather than 95% values, etc., the uncertainty values are not unrealistic.

3. All references of the design of other B&W NSSS plant designs should be deleted. We have no current knowledge of the present design of any other plant and to state that a particular Oconee feature is or is not part of another plant's design is not a statement of fact.

Response: We would agree that the conclusions, mitigating actions, etc. should be discussed with respect to Oconee since that is the plant from which the data was obtained. Inference of conditions at other B&W plants based on Oconee results without a review of the other plants is inappropriate. References to B&W plants in general used in this context were changed to refer only to Oconee.

4. Repeated reference is made to the best estimate TWC value of 4.5E-6/RY. It should be made clear that this value only applies at 32 EFPY and that prior to that time the TWC probability is lower.

Response: We agree with the comment. The 32 EFPY designation was added to the discussions of the best estimate TWC value.

Chapter 1.0 Comments

5. The statement regarding future use of the Oconee PTS model in the first paragraph is inappropriate. It serves no useful purpose, and is contrary to the implied agreement we had. Duke requests that it be deleted.

Response: The sentence was deleted from the text.

Chapter 2.0 Comments

6. The vessel downcomer is between the vessel and the thermal shield, not the core barrel.

Response: We concur. The change was made in the text.

7. The coolant inventory control system adds or withdraws coolant based on indicated inventory, not reactor power.

Response: We concur. The change was made in the text.

8. It should be made clear that safety injection is used to prevent fuel overheating only under accident conditions.

Response: We agree. The text was changed to make this point.

9. The condensate train is incorrectly described in the second paragraph. The condensate path is F Heaters, E Heaters, D Heaters, C Heater Drain Coolers, and C Heaters. The D Heaters are not drain coolers.

Response: We concur. Text was revised to reflect this comment.

10. The condensate booster pumps are capable of delivering feedwater at 550 psig. not 600 psig.

Response: We agree. The change was made in the text.

11. The EFW pumps are actuated if the MFW pumps trip off or if MFW pressure drops below 750 psig.

Response: We agree. The change was made in the text.

12. Section 2.2.3.3 is completely erroneous. ICS no longer controls EFW. EFW is completely separate from ICS and is initiated and controlled by a safety-related system.

Response: We agree. The reference to the ICS system was in error. This reference was removed from the text.

13. In Section 2.2.4, no basis has been provided to support the statement that there is a historical record of single failures of support systems causing overcooling transients.

Response: We agree. The text was changed to reflect the potential importance of support systems rather than any historical data.

14. Secondary side pressure should be 1010 not 1025 psig.

Response: We agree. Change in the text was made.

15. It is asserted that LOCAs can generate stagnated flow regimes that can contribute to "localized rapid cooldown of the reactor vessel." Yet, in the last sentence of this section it is stated "the vent valves, therefore, promote continued circulation of coolant through the core region as well as provide warm water to mix with HPI flows entering the downcomer region." This latter discussion is representative of the true situation and would seem to contradict the first sentence.

Response: The first sentence was meant to imply that in general LOCAs have a potential to generate stagnated flow regimes that can contribute to localized rapid cooldown of a reactor vessel. The rest of the discussion goes on to show that at Oconee the operation of the vent valves promotes circulation and therefore precludes or greatly limits the potential for stagnated flow regimes. Some rewording of this section was made to avoid misunderstandings.

Chapter 3 Comments

16. There are two TSVs on each branch, not four.

Response: We agree. The change was made in the text.

17. The report states that secondary upsets will tend to mask a PORV LOCA, and goes on to assume that the operators will do nothing to mitigate such an event until steam generator heat removal capacity is restored. This is clearly inconsistent with post-TMI operator training

programs. We do not believe this takes into account the addition of subcooling margin monitors and core exit thermocouples to the control rooms.

Response: We concur that the assumption made was a conservative approach and the report was changed to make this point.

Chapter 4 Comments

18. The emergency feedwater system control is not part of the ICS.

Response: The statement as presented in the text is in reference to the overall control system not just the ICS. However to avoid confusion additional wording was added to clearly indicate that the EFW is not part of the ICS.

19. The large uncertainties should have no effect on the best estimate value. All they can do is affect overall uncertainty.

Response: We agree. The best estimate value should not have been changed. Thus, the use of the adjusted thermal-hydraulic transient is inconsistent with the treatment of uncertainty later in the report. The result of this inconsistency is a conservative bias for this transient. A sentence was added in the text to make this point.

20. The sequence of events incorrectly indicates that EFW is being started on low SG level.

Response: We agree. The EFW flow initiates at the time indicated. However, the pumps had previously started. Reference to the pumps starting at indicated time was removed.

21. For failure of two turbine bypass valves at full power other significant features of the transient include: Operator does not isolate/control feedwater, high level trip fails, MFW is not realigned upon RCP trip.

Response: We would agree that "operator does not isolate/control feedwater" is a feature which should have been identified. This feature was included in the discussion of the transient. The high level trip was not assumed to fail and in fact tripped the pumps at 60.7 seconds as indicated in table 4.15. Furthermore, MFW was realigned upon RCP trip as indicated in table 4.15.

22. The sequence of events shows MFW pump trip at 61 seconds, but no EFW actuation until 210 seconds. This does not make sense.

Response: The reference to pump actuation at 209 seconds was meant to refer to actual EFW flow with pumps having been actuated following MFW pump trip. The text was changed to make this point.

23. The extrapolations of temperature for several of the sequences are unrealistic in a physical sense and are substantially conservative.

Response: We would agree that the extrapolated temperatures appear to be conservative. However, in light of the potential cooling of the HPI and EFW systems the level of conservatism could not be determined without a thermal-hydraulic calculation of the complete two hour analysis period. Since this was not done, the curves presented in the report do reflect our best estimate based on the available data and thus no change was made in the discussion of the thermal-hydraulics associated with these sequences.

24. There were several general comments made by Duke Power Corporation concerning what they perceived to be considerable conservatisms used in the extrapolation process.

Response: Although ORNL intuitively agreed with many of these comments, changes in the report could not be made without specific data. This data could not be obtained without a great deal of further thermal-hydraulic calculations. As a result, the data as presented in chapter 4 represents the only data available and thus must be used as best estimate information.

25. Reference is made to a study "under investigation". The specifics should be provided as a footnote or reference.

Response: The reference was made to research being performed at Pacific Northwest Laboratories. It was decided, however, that reference to this study detracted from the point being made in the text and thus the reference was deleted.

26. It seems that substantial unjustified uncertainty is being applied to flaw density distribution.

Response: The flaw density distribution is based upon the engineering judgement of our expert in fracture mechanics. The distribution used was not intended as a typical distribution for a particular reactor vessel. Instead, it should be interpreted as covering the population of all possible reactor vessels. However, the fact that a particular ISI detected no near surface flaws of concern to PTS does not necessarily reduce the uncertainty in the flaw density distribution. The ISI techniques used are probably not capable of detecting flaws in and extending a short distance beyond the cladding. Thus no changes in the text were made.

27. For the data provided in Appendix E, insufficient definition of terms is provided.

Response: We agree that in some instances definition of terms was not provided. Information was added for those terms not previously defined in the text.

28. Replacement of the current HPI system with a low head system is discussed. This item indicates a limited knowledge of the big picture of plant safety and should be deleted.

Response: The comment was not meant to be interpreted as a recommenda-

tion for a change in the Oconee design. The purpose of the discussion was to indicate the importance of the repressurization process to the PTS phenomenon. The text was changed to make this point clear.

29. The discussion of ISI in this chapter completely ignores the fact that such inspections are regulatory requirements.

Response: The inservice inspections discussed in this chapter are different than the inservice inspections presently required by NRC. It was clear that this distinction was not made. The text was changed to clearly point out that we were referring to improved inservice inspection techniques.

30. Thermal-hydraulic distributions were assumed to be normal, yet the dominant contributors, as previously noted, were conservatively predicted as either upper pressure or lower temperature bound. It is simply incredible that a transient with an assumed T^f of below 212° can have an uncertainty that would put the value even lower.

Response: Even though it appears that conservatisms were included in the thermal hydraulic analyses, it was necessary to assume a normal distribution for the thermal-hydraulic variables since only limited information was available. It should be noted that although these conservatisms may be large, they should not be interpreted as absolute upper and lower bounds of pressure and temperature. Finally, a significant importance is implied to the 212° temperature. This does not represent a bound for the steam line break events except in the case where steam line break occurs with short duration of HPI and EFW flow. However, we would agree that as these lower temperature values are reached the uncertainty in the temperature values should significantly decrease.

31. It is incredible that in light of the ISI that have been done, the flaw density could be substantially higher than that conservatively assumed.

Response: The text was changed to emphasize the opinion that the present ISI techniques in use are not sufficient to detect the types of flaws which may be important to PTS.

Chapter 8 Comments

32. The conclusions imply that PTS is a safety concern, that there is a high degree of uncertainty in the non-conservative direction and that further study is required in several areas.

Response: The conclusions were not meant to imply that PTS is a safety concern at Oconee. In fact the opposite is stated in the conclusions secitions. We would agree that the need for further study of certain areas of the analysis may have been overstated. As a result some of the references to such studies were removed. We would still imply that there are uncertainties in the results. Although we would intuitively agree that the uncertainty may be somewhat high in the non-conservative direction due to conservative assumptions made in the analysis, there is

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not enough sufficient data to imply the level of conservatism. In light of this, no changes were made in the uncertainty values.

33. The proposal to heat HPI by 50°F contradicts a statement made in Section 6.3.4: "Overall, HPI heating could provide some small but variable benefit in terms of downcomer temperature response".

Response: The wording in this section was misleading. The text was changed to make the point clearer. It is our opinion that there is not a contradiction.

34. The Anticipated Transient Operating Guidelines (ATOG) program now being implemented at B&W plants includes operator actions to limit primary system repressurization as part of the response and mitigation of overcooling events. This should be acknowledged in the NUREG if the recommendation is to be retained.

Response: Reference to the ATOG program was made.

35. Section 8.2.5 item (4) provides the uncertainty distribution and states that the Oconee PTS risk is four times higher than the safety goal. This should be revised to show the integral relationship with the safety goal using the point estimate with and without the residual.

Response: After careful consideration, it was ORNL's decision to remove the discussion of the comparison of the 95% value and the safety goal. It was our opinion that there is conservatism in the 95% confidence level. Therefore, actual comparisons of the 95% value and the safety goal may be misleading. As a result, item (4) was deleted from the conclusions section.

36. In the summary section of chapter 8, the first sentence is inappropriate and should be removed. It was not the stated purpose of this report to judge the final status of pressurized thermal shock as a safety issue.

Response: A review of the referenced sentence, indicated that conclusions were implied which were not justified. The sentence was deleted from the text.

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13 ABSTRACT (200 words or less)

An evaluation of the risk of pressurized thermal shock (PTS) resulting in a through-the-wall crack in a reactor pressure vessel was performed for the Oconee Unit 1 nuclear power plant. The information presented in this report covers one of three plant-specific studies performed for NRC. The other two studies, for Calvert Cliffs Unit 1 and H. B. Robinson Unit 2, are documented in NUREG/CR-4022 and NUREG/CR-4183, respectively. The specific objectives of the Oconee study were (1) to develop a basic methodology for evaluating the risk of PTS that could be refined later in the other two studies, (2) to provide a best estimate of the frequency of a through-the-wall crack for the Oconee Unit 1 vessel, (3) to determine the dominant PTS sequences for the unit, and (4) to evaluate the effectiveness of potential corrective measures. The examination of tens of thousands of transients indicated that PTS was not an important core melt initiator for Oconee Unit 1. The dominant risk sequences were determined to be large steam-line break events.

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